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Title: MCNP6 Delayed Neutron Emission Validation with
Experimental Measurements

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Intended for:



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MCNP6 Delayed Neutron Emission Validation with Experimental Measurements

Presented by:
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Supervisors:
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Dr. Emily Corcoran, RMC
Dr. David Kelly, RMC
LA-UR 11-05868

October 11 2011

Overview

- Introduction, Background & Theory
- Experimentation
- MCNP6 Model
- Results & Discussion
- Conclusions

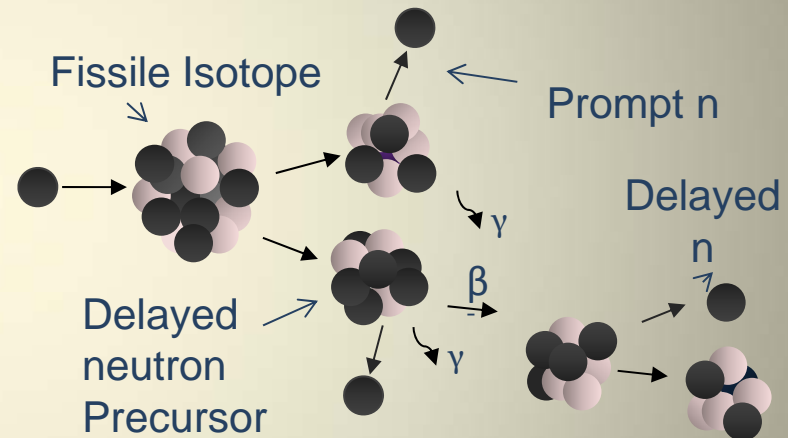
The Delayed Neutron Counting System

- Constructed in 2010
- Validated for ^{235}U analysis, aqueous samples
- Current experiments include ^{233}U samples
- Expanding analysis to include ^{239}Pu analysis and mixtures of two or more fissile isotopes
- Winter 2011/2012 will be updated with new hardware to increase sensitivity and efficiency

Delayed Neutron Generation

- Many delayed neutron precursors

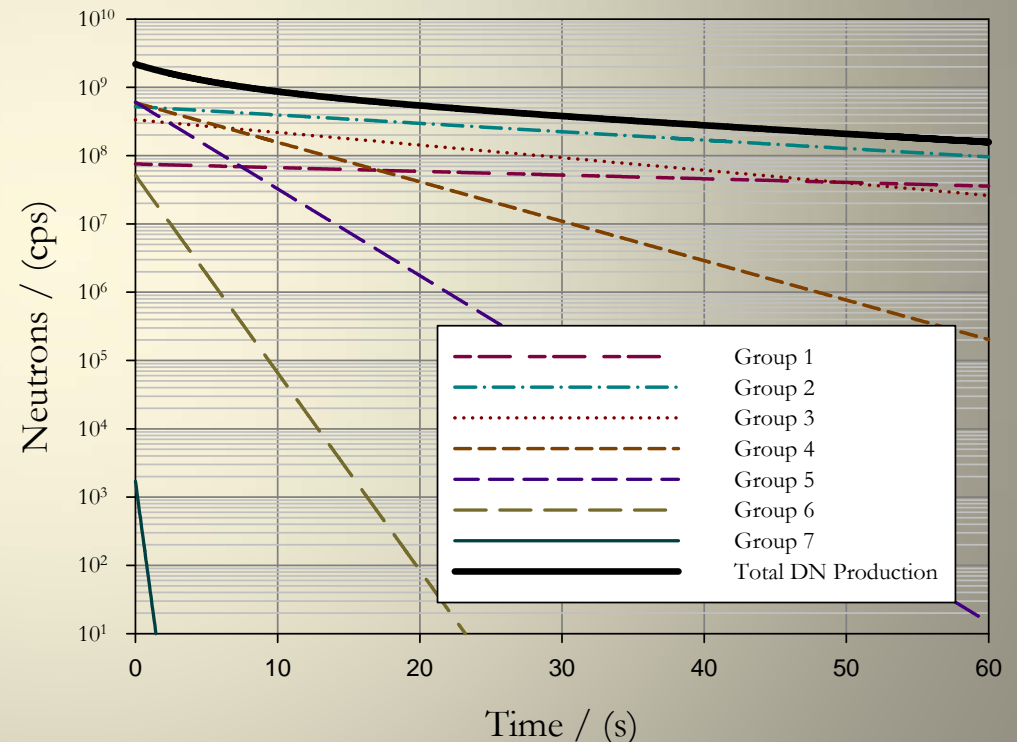
Group	$t_{1/2}$ [s]	λ [s^{-1}]	$\beta_i = \nu_i/\nu_d$ [%]
1	55.6	0.014267	0.0328 ± 0.0042
2	24.5	0.028292	0.1539 ± 0.0068
3	16.3	0.042524	0.091 ± 0.009
4	5.21	0.133042	0.197 ± 0.023
5	2.37	0.292467	0.3308 ± 0.0066
6	1.04	0.666488	0.0906 ± 0.0046
7	0.424	1.634781	0.0812 ± 0.0016
8	0.198	3.554600	0.0229 ± 0.0095



Delayed Neutron Generation

- Eight (or 6) groups, denoted by $t_{1/2}$ and production ratio

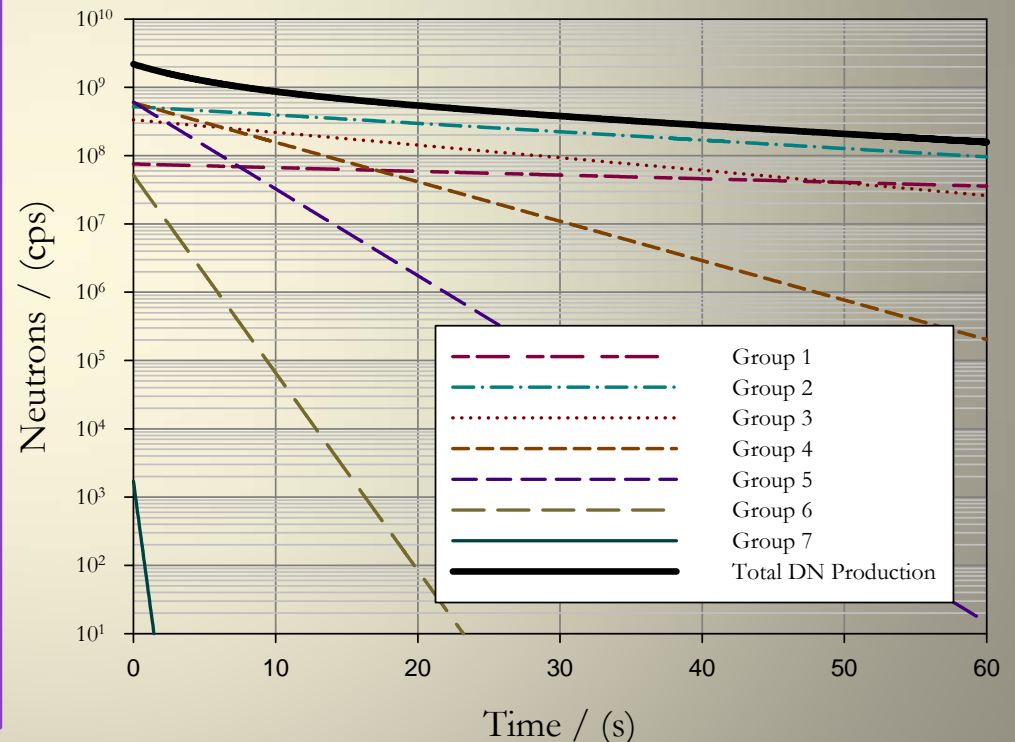
<i>Group</i>	$t_{1/2}$ [s]	λ [s^{-1}]	$\beta_i = \nu/\nu_d$ [%]
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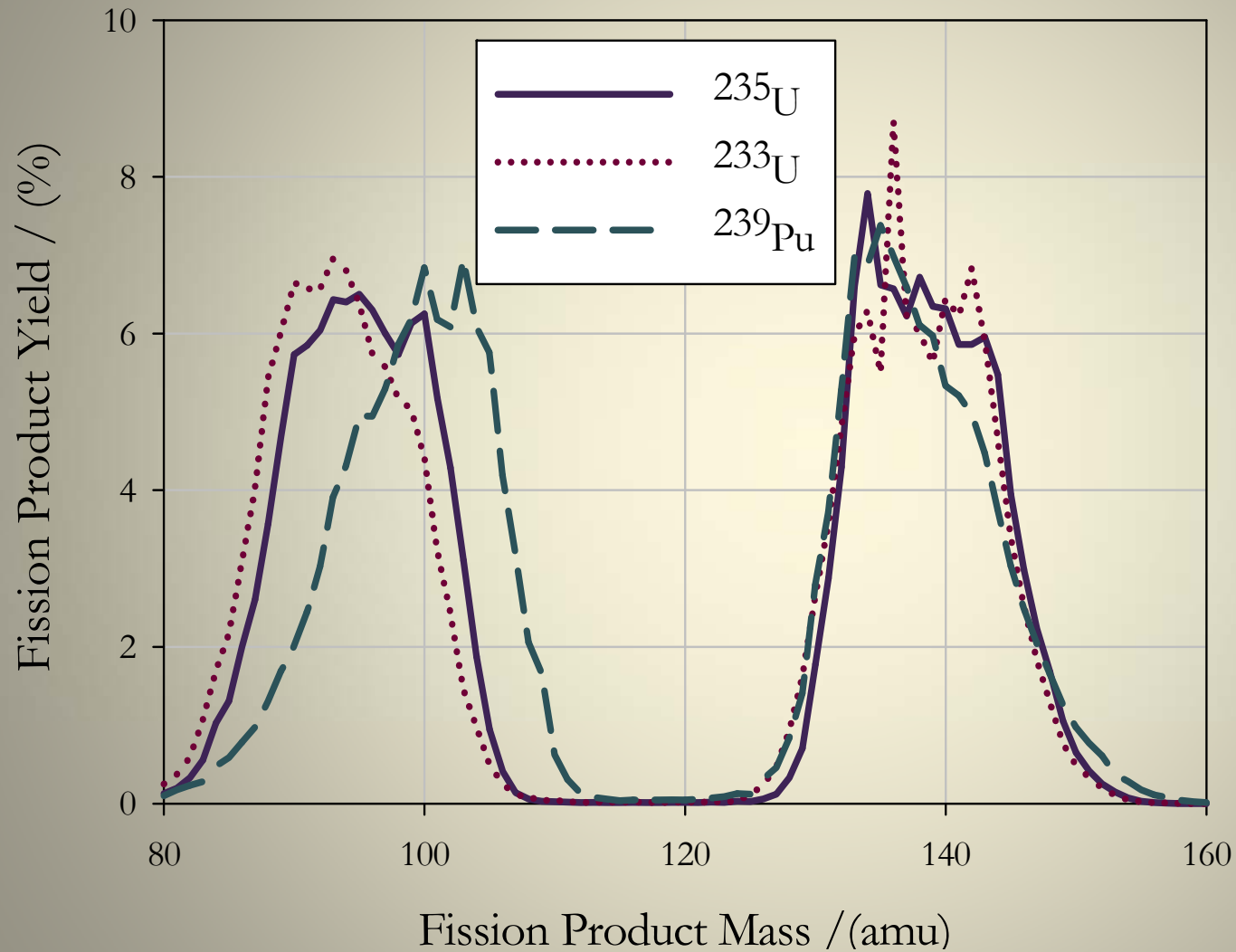


Delayed Neutron Generation

- Production ratios dictated by fission fragment yields

<i>Group</i>	$t_{1/2}$ [s]	λ [s^{-1}]	$\beta_i = \nu/\nu_d$ [%]
1	55.6	0.014267	0.0328 ± 0.0042
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$$\beta_i = \nu_i / \nu_d [\%]$$

$$0.0328 \pm 0.0042$$

$$0.1539 \pm 0.0068$$

$$0.091 \pm 0.009$$

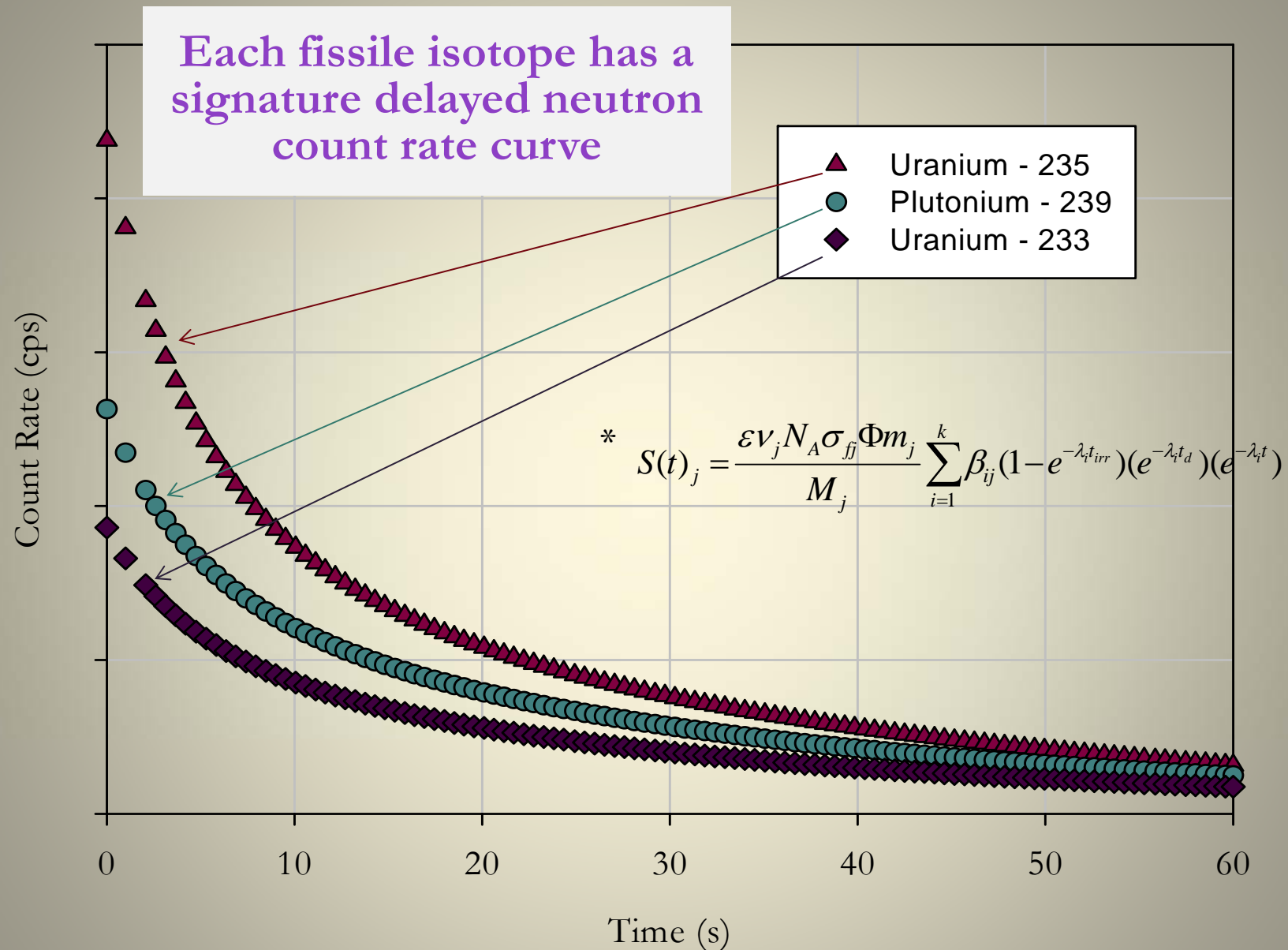
$$0.197 \pm 0.023$$

$$0.3308 \pm 0.0066$$

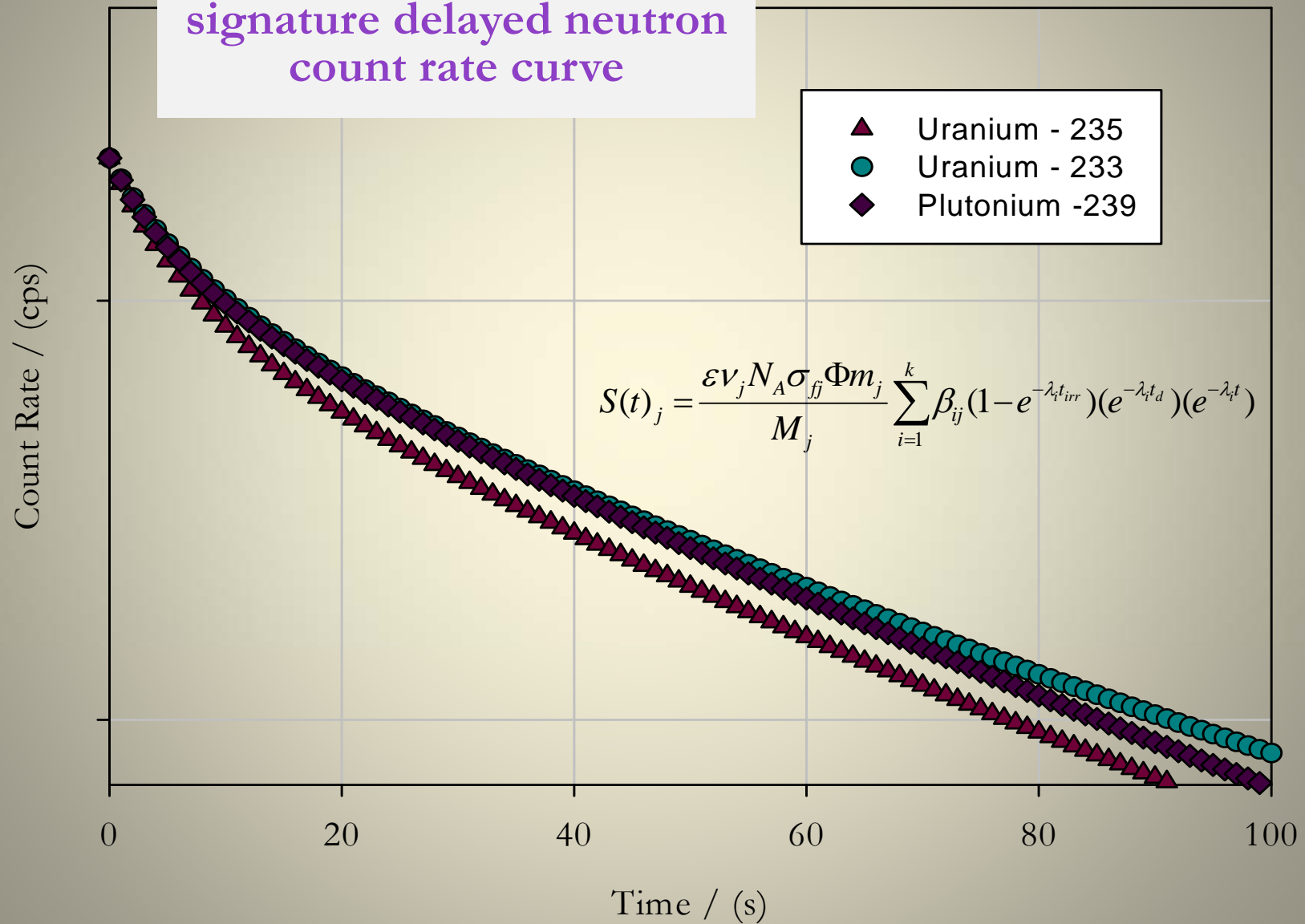
$$0.0906 \pm 0.0046$$

$$0.0812 \pm 0.0016$$

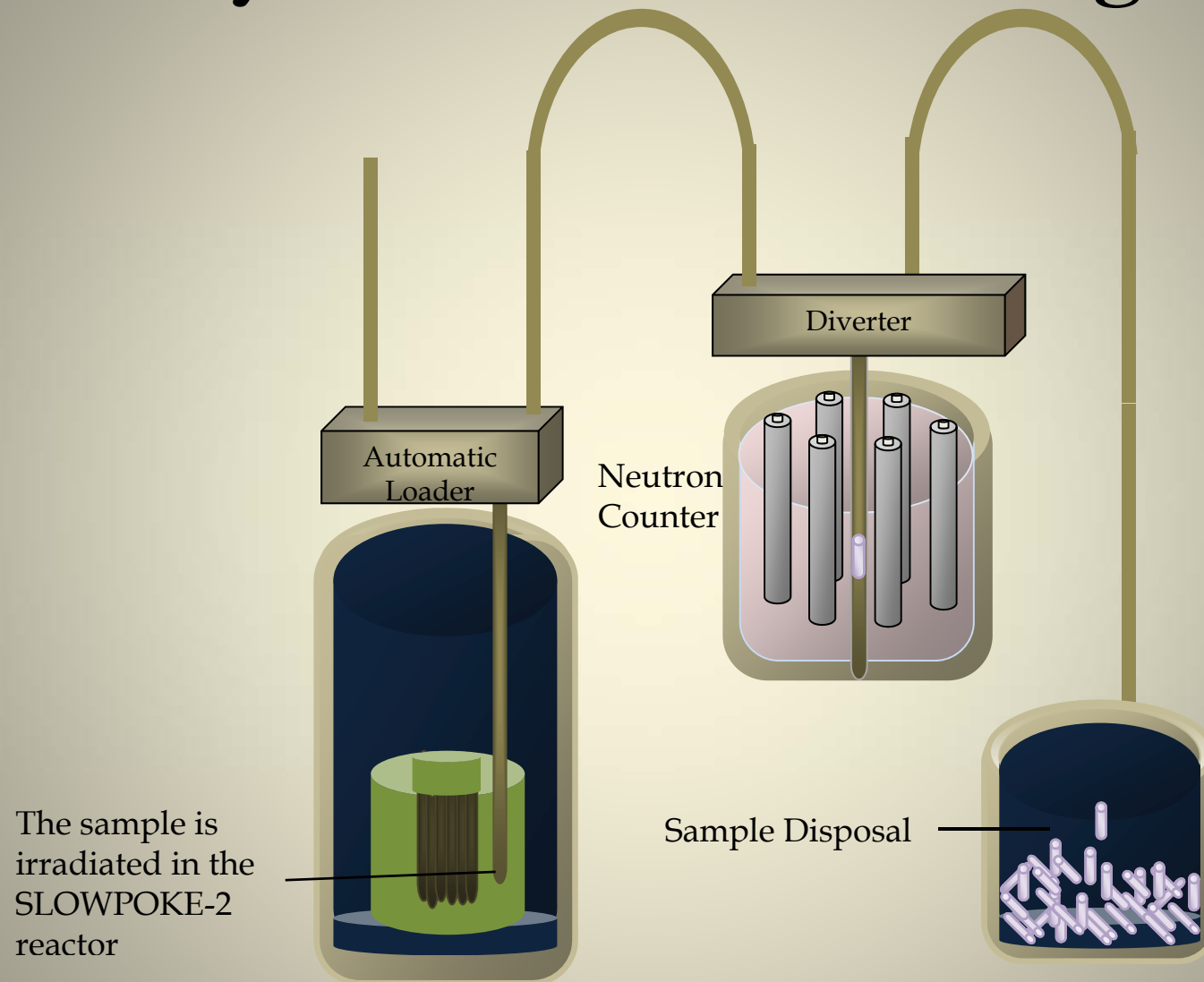
$$0.0229 \pm 0.0095$$



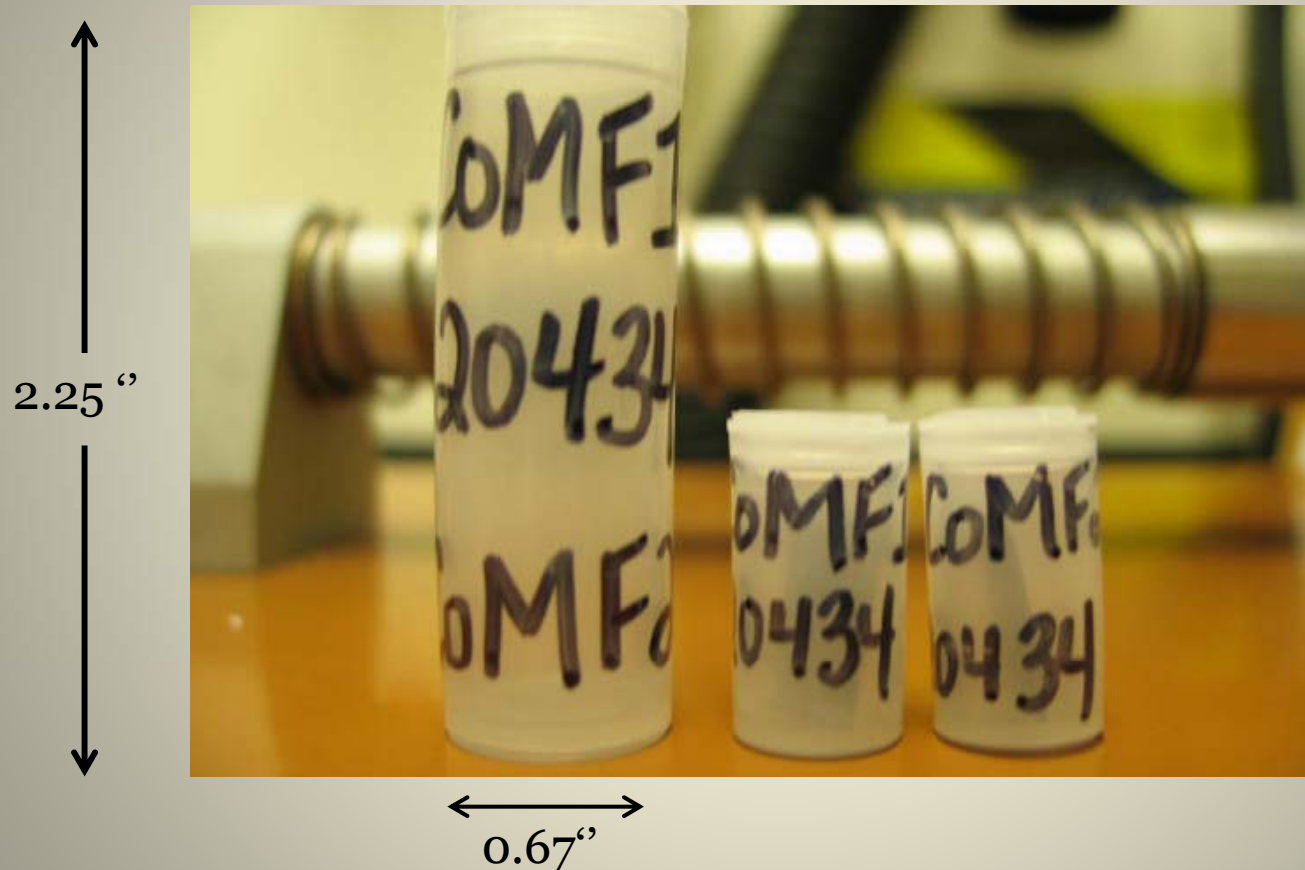
Each fissile isotope has a signature delayed neutron count rate curve



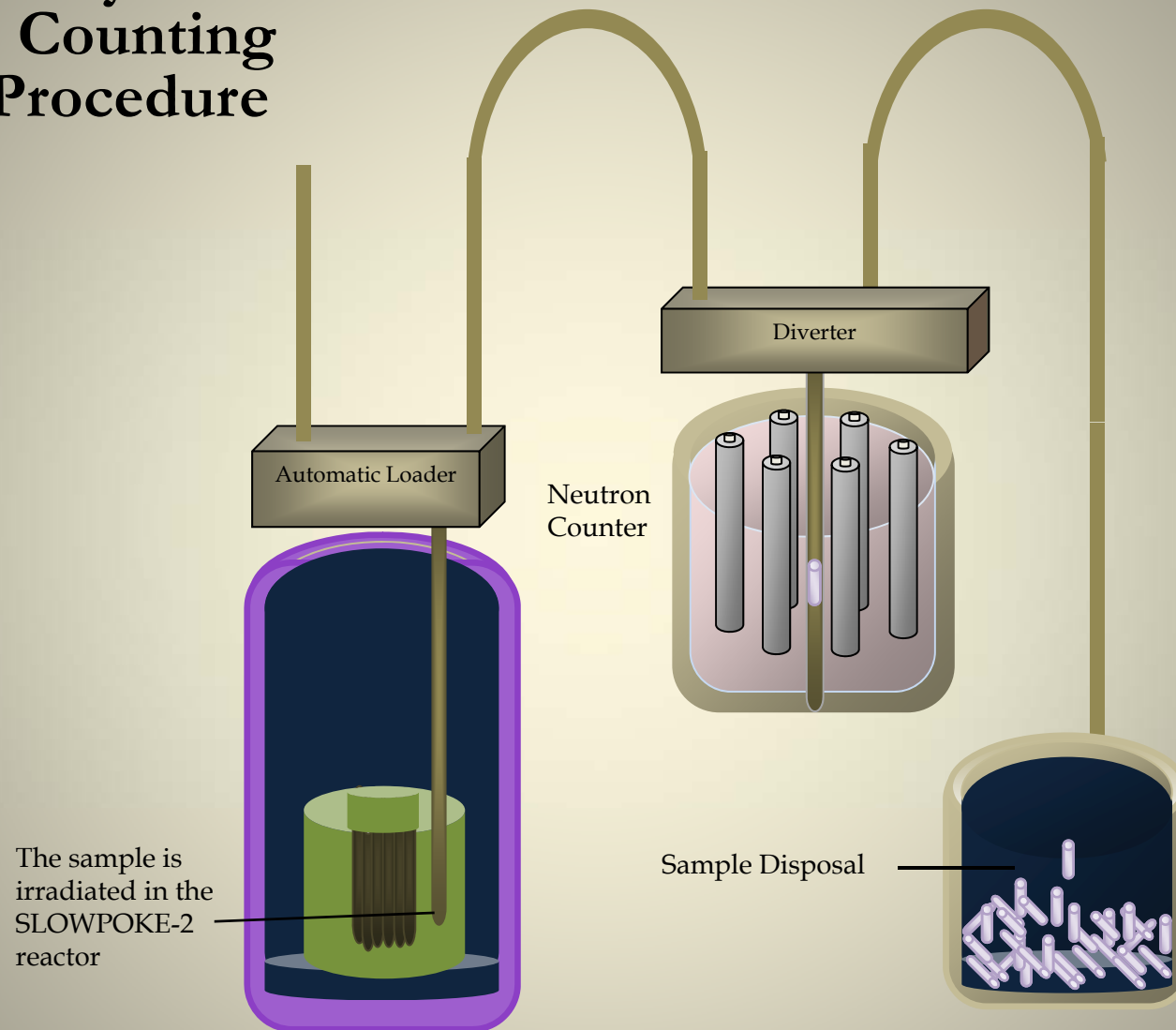
The Delayed Neutron Counting System



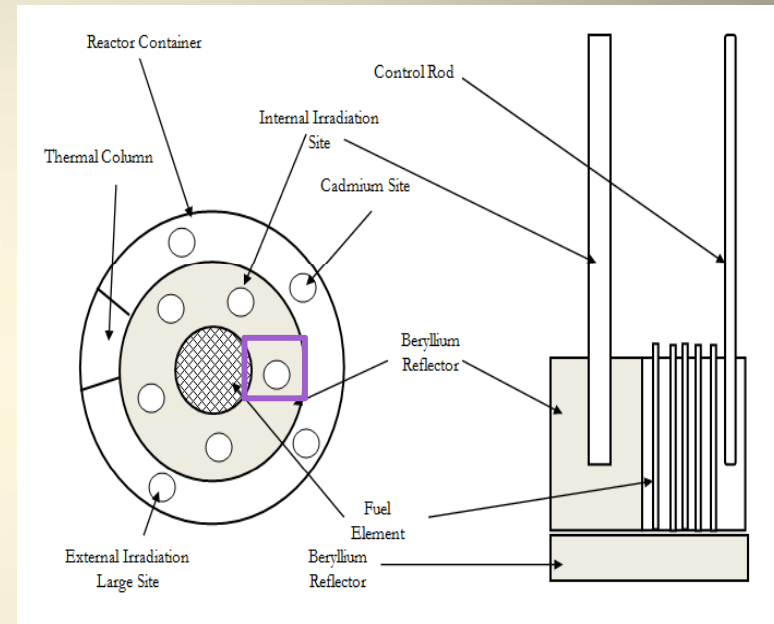
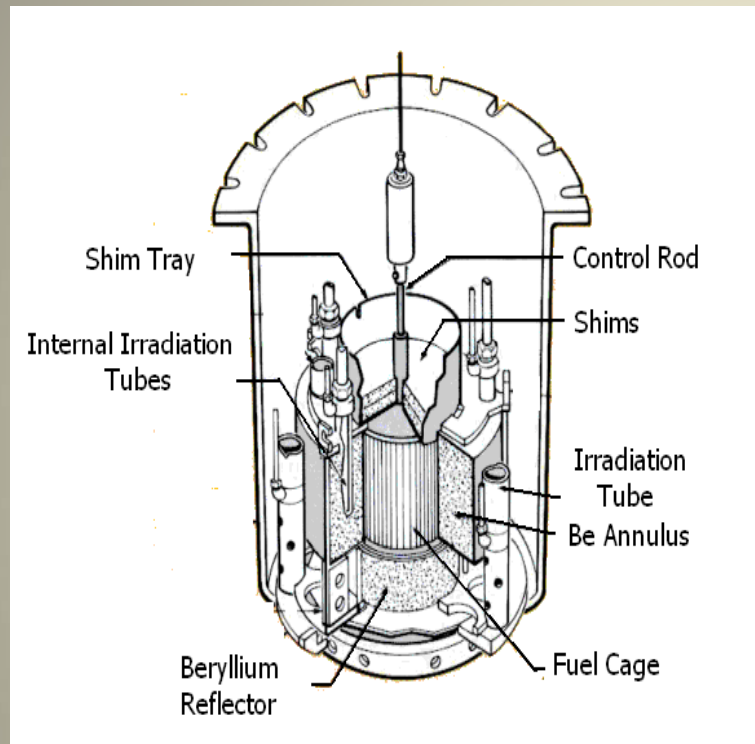
Sample Preparation



The Delayed Neutron Counting System Procedure

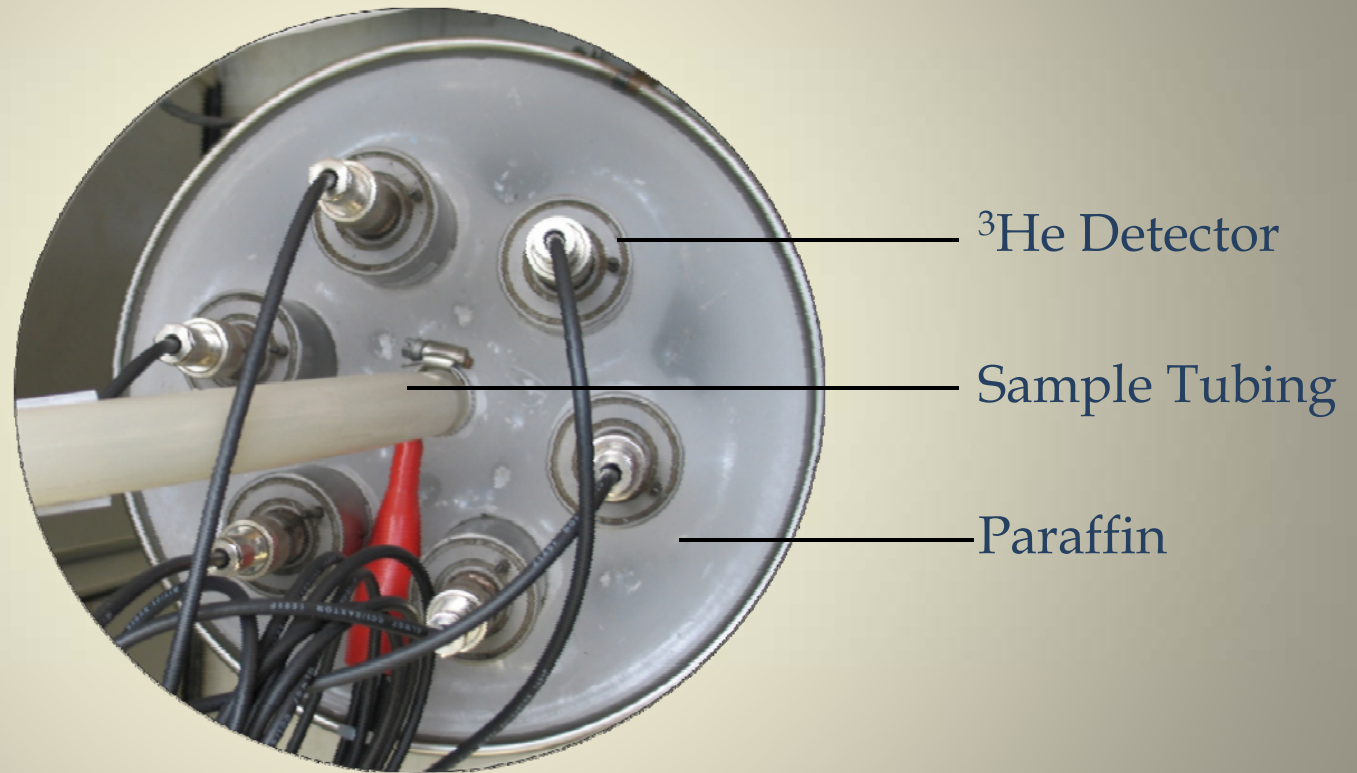


The SLOWPOKE-2 Reactor

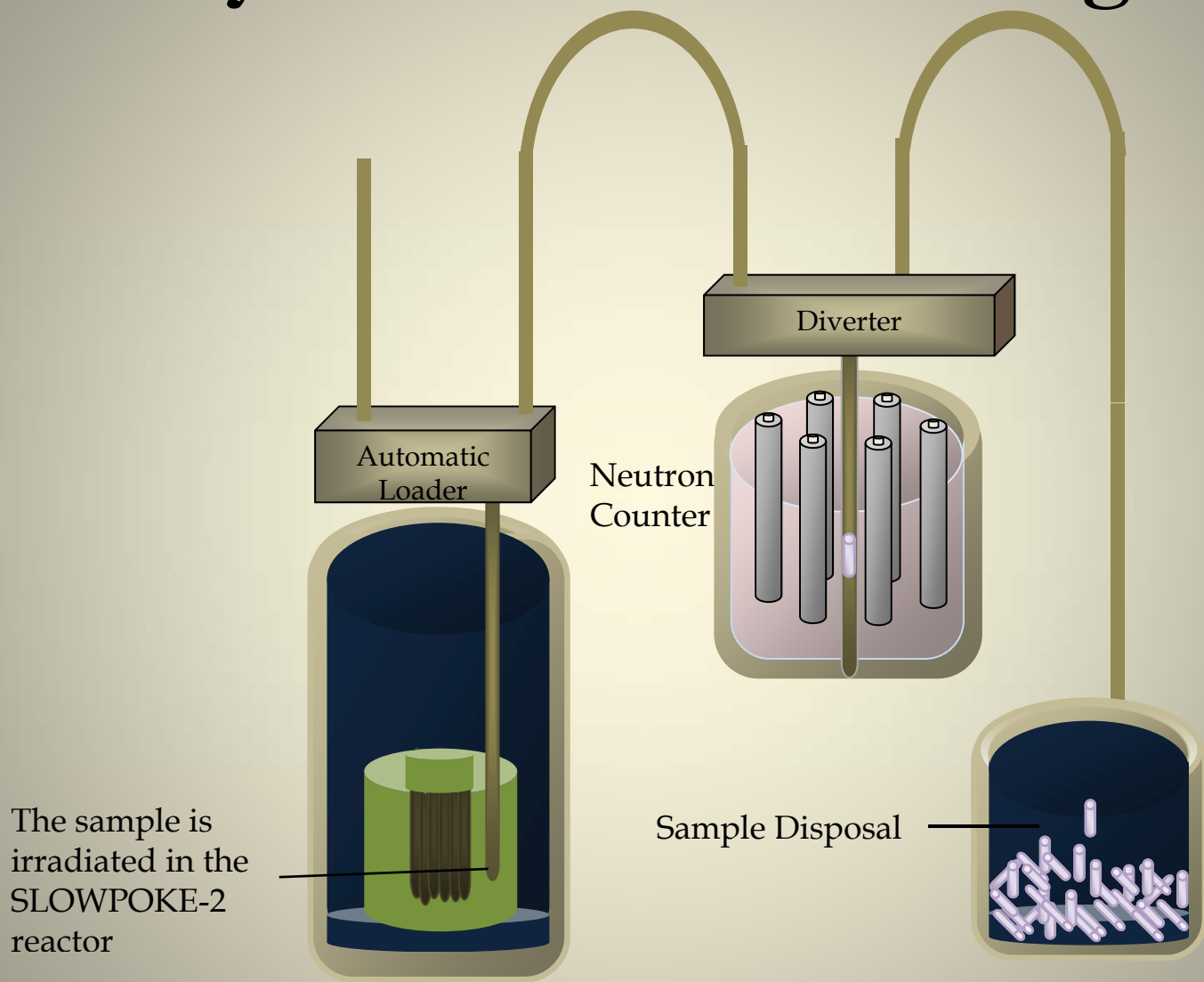


20 kW research reactor enriched to 19.89%

Delayed Neutron Counting Arrangement



The Delayed Neutron Counting System



Hardware & Software Control

The screenshot displays the software control interface for the MCNP6 Model of DNC System. The interface is divided into several sections:

- Experimental Parameters:** A panel on the left containing various input fields and controls. It includes a "STOP PROGRAM" button and a "Select MCA Output" button with a red error message "Control could not be loaded".
- Central Diagram:** A flow diagram showing the process flow between four main components: "Automated Sample Loader", "Irradiation Site", "Delayed Neutron Counter", and "Disposal Unit". A "Sample" field is set to 3 and a "Cycle" field is set to 1. A central diagram shows a sample being loaded into an irradiation site, then moving to a delayed neutron counter, and finally to a disposal unit. A chemical diagram of a nuclear reaction is also visible.
- Valve Indication:** A section at the bottom left showing five valves (Valve 1, Valve 2A, Valve 2B, Valve 3, Valve 4, Valve 5) with their respective status indicators (green or grey) and a "Reset Photo Latch" button.
- Manual Control Panel:** A section at the bottom right with five physical-style buttons labeled "Valve 1", "Valve 2A", "Valve 2B", "Valve 3", "Valve 4", "Valve 5", and "Reset", along with a "QUIT" button.

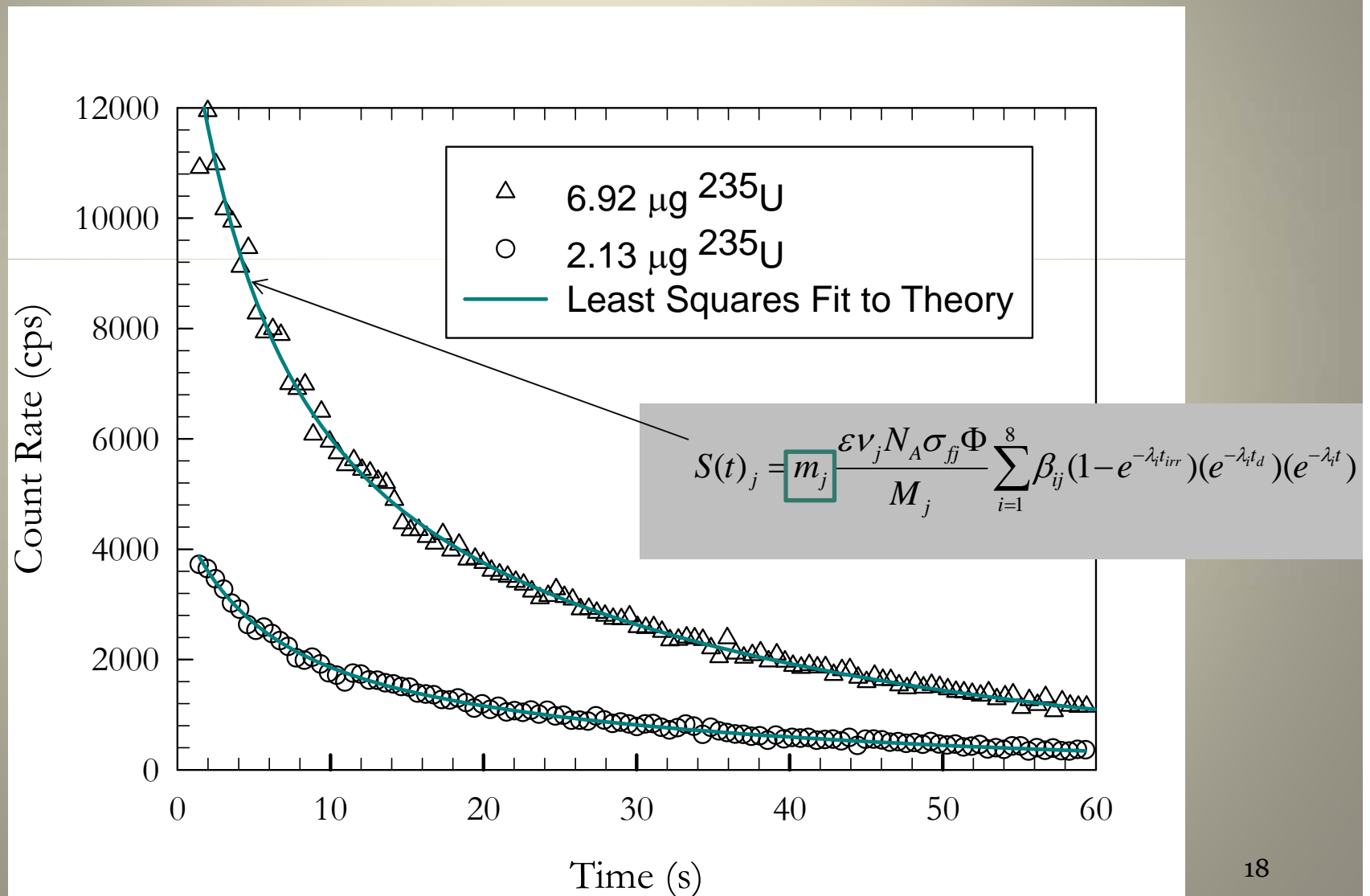
The Fissile Analysis Program

- Imports count excel file:

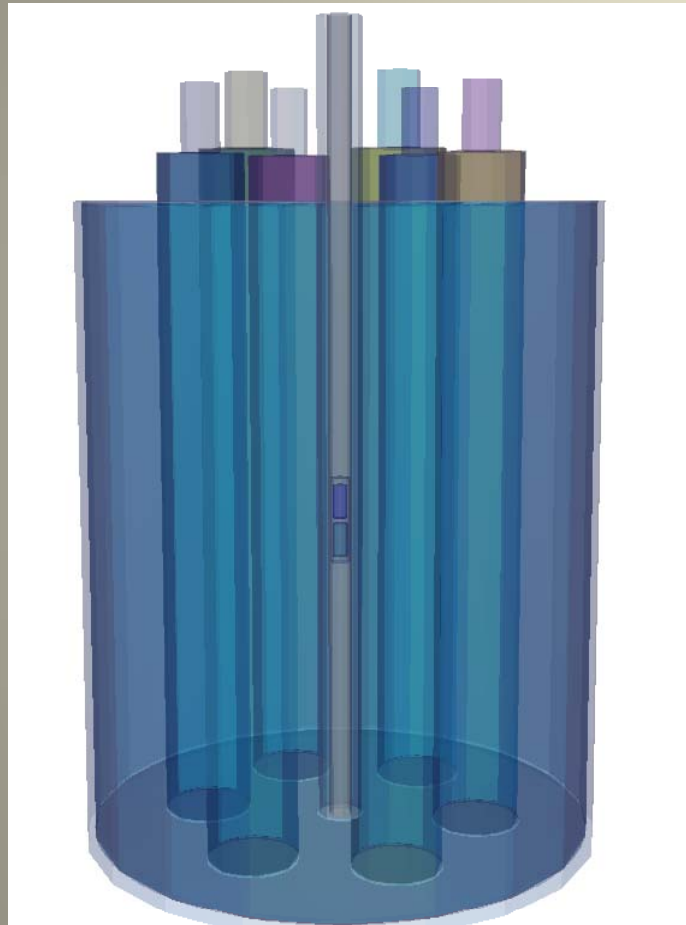
SLOWPOKE Test Data				
11/04/2011	11:47:49 AM			
Standard? YES				
Sample #	Cycle	Time	Counts	Total Counts
1	A	1.015625	1863	79306
		1.546875	1821	
		2.0625	1759	
		2.59375	1691	

- Corrects for background, dead time, normalizes to counts per second
- Outputs fissile (U-235) content in μg

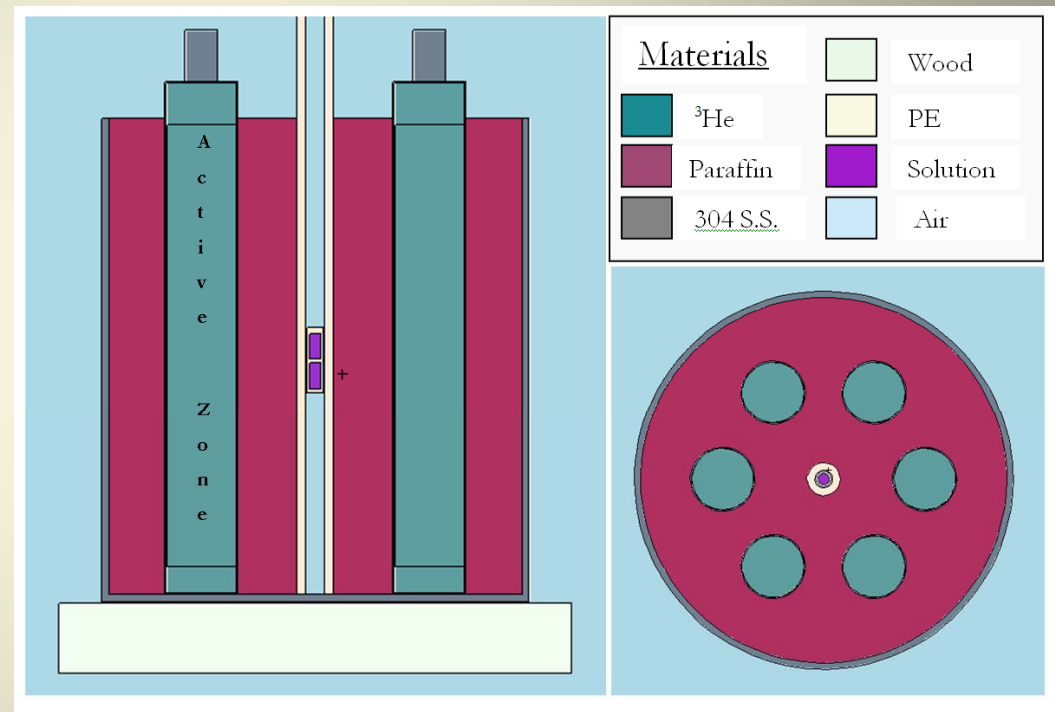
Fissile Analysis Program Output



MCNP6 Geometry

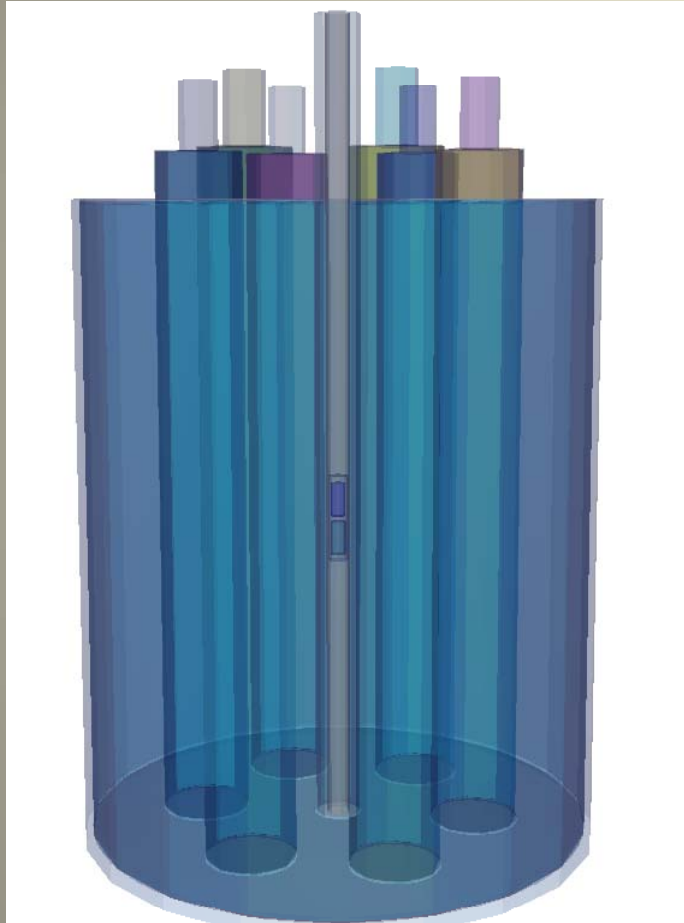


Visual Editor



MCNP6 Plotter

MCNP6 Input Deck Summary



Visual Editor

- $tme = 0 - 60e8$ sh
 - Sample is exposed to defined neutron flux
 - DN activity builds
- $tme = 60e8 - 180e8$ sh
 - F8 tallies record delayed neutron activity

Slide 20

MS8

Describe the INPUT DECK

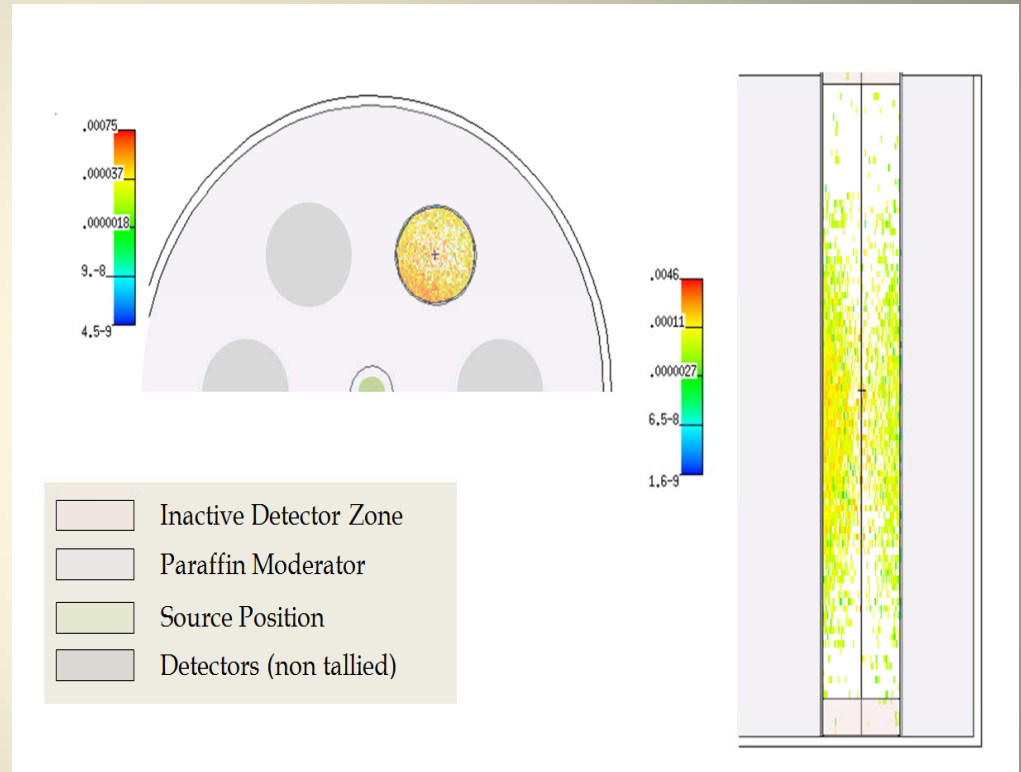
Madison, 10/11/2011

^3He Detectors & System Efficiency

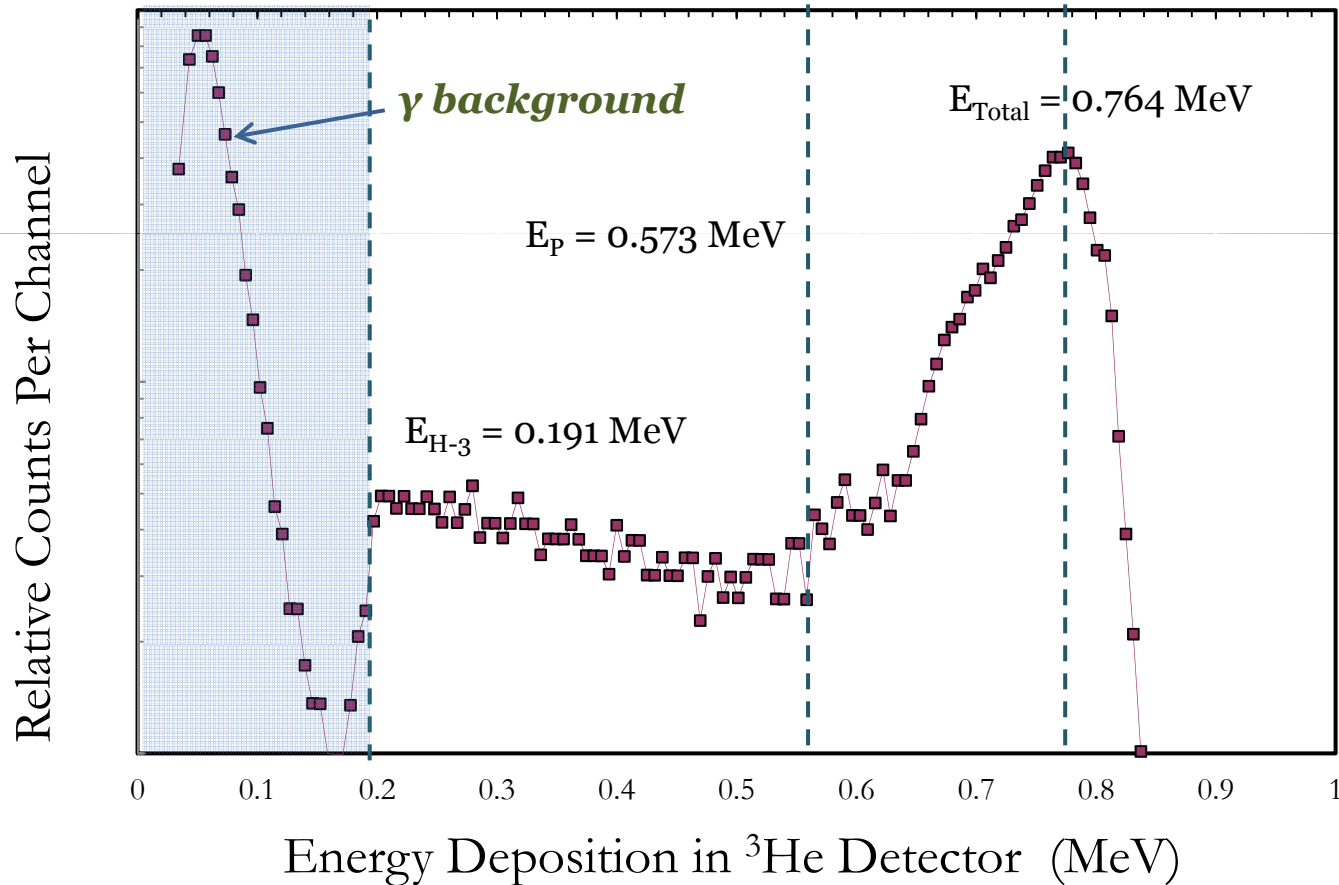
- F8 tallies (pulses) in active zone of detector
- User-defined delayed neutron energies and temporal behaviour using 8-group model

$$S(t)_j = m_j \frac{\varepsilon \nu_j N_A \sigma_{ff} \Phi}{M_j} \sum_{i=1}^8 \beta_{ij} (1 - e^{-\lambda_i t_{irr}}) (e^{-\lambda_i t_d}) (e^{-\lambda_i t})$$

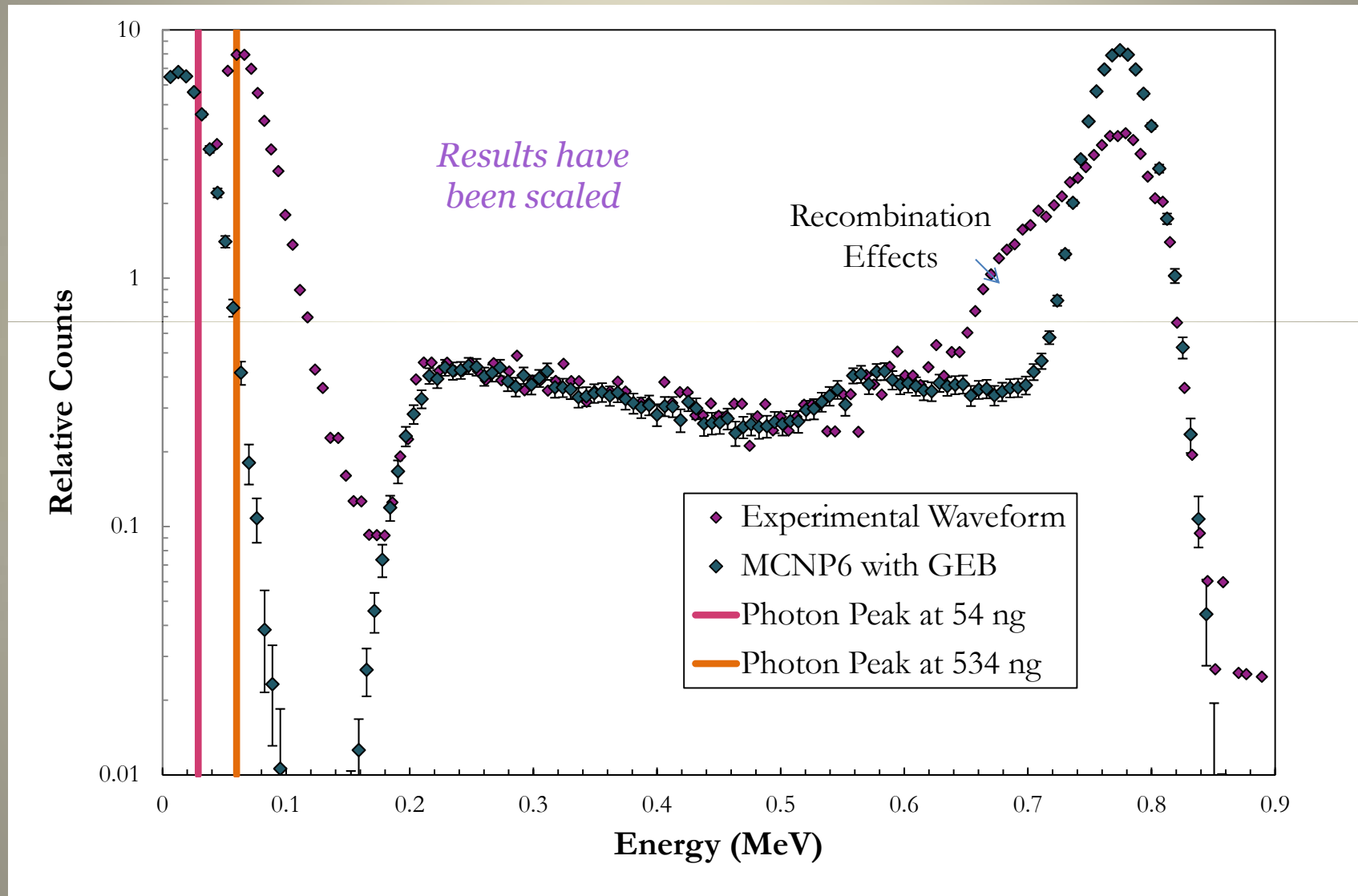
- Experimental Efficiency: $34 \pm 5\%$
- MCNP6 Efficiency: 37%



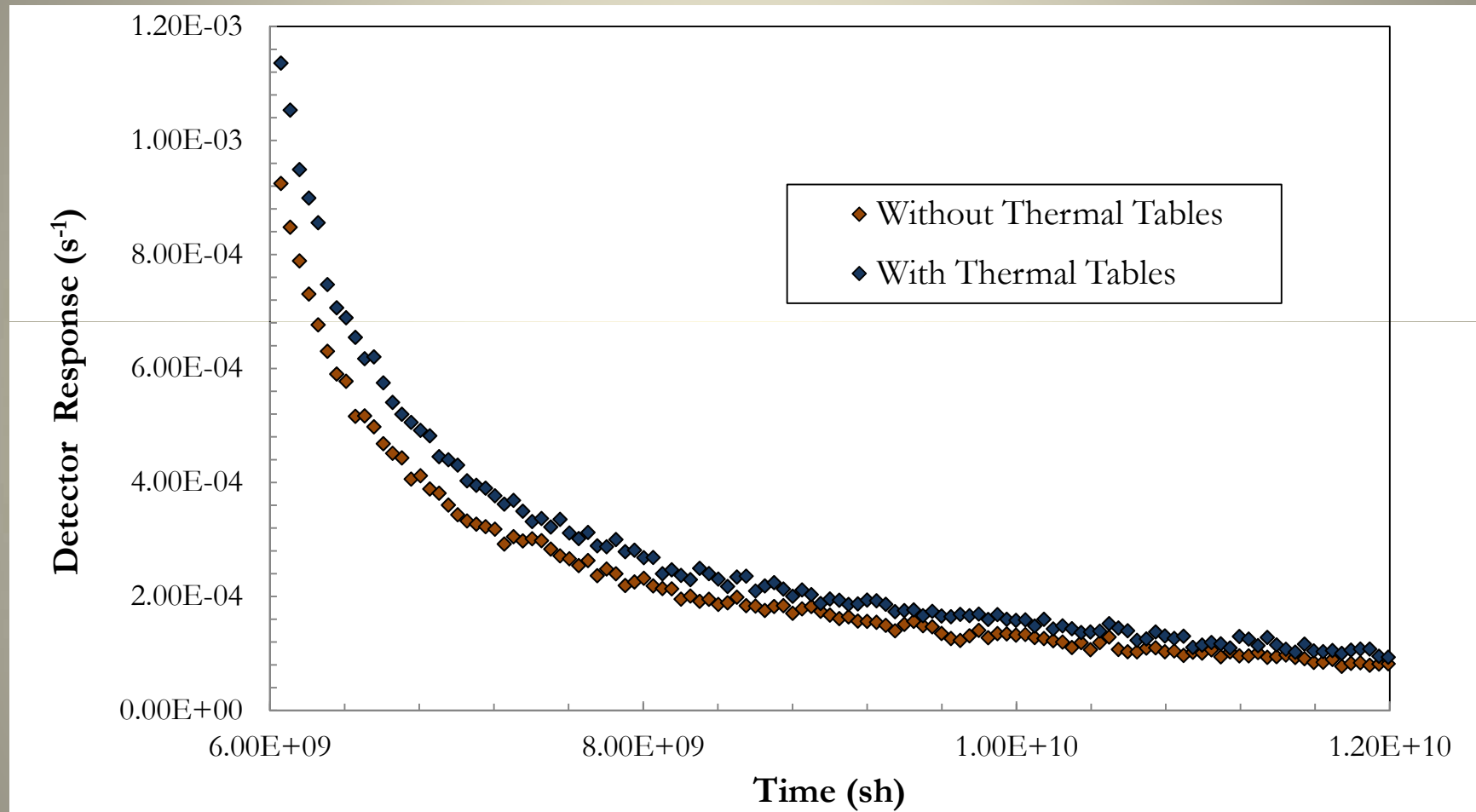
Detector Wall Effects – Experimental Measurements



Detector Wall Effects

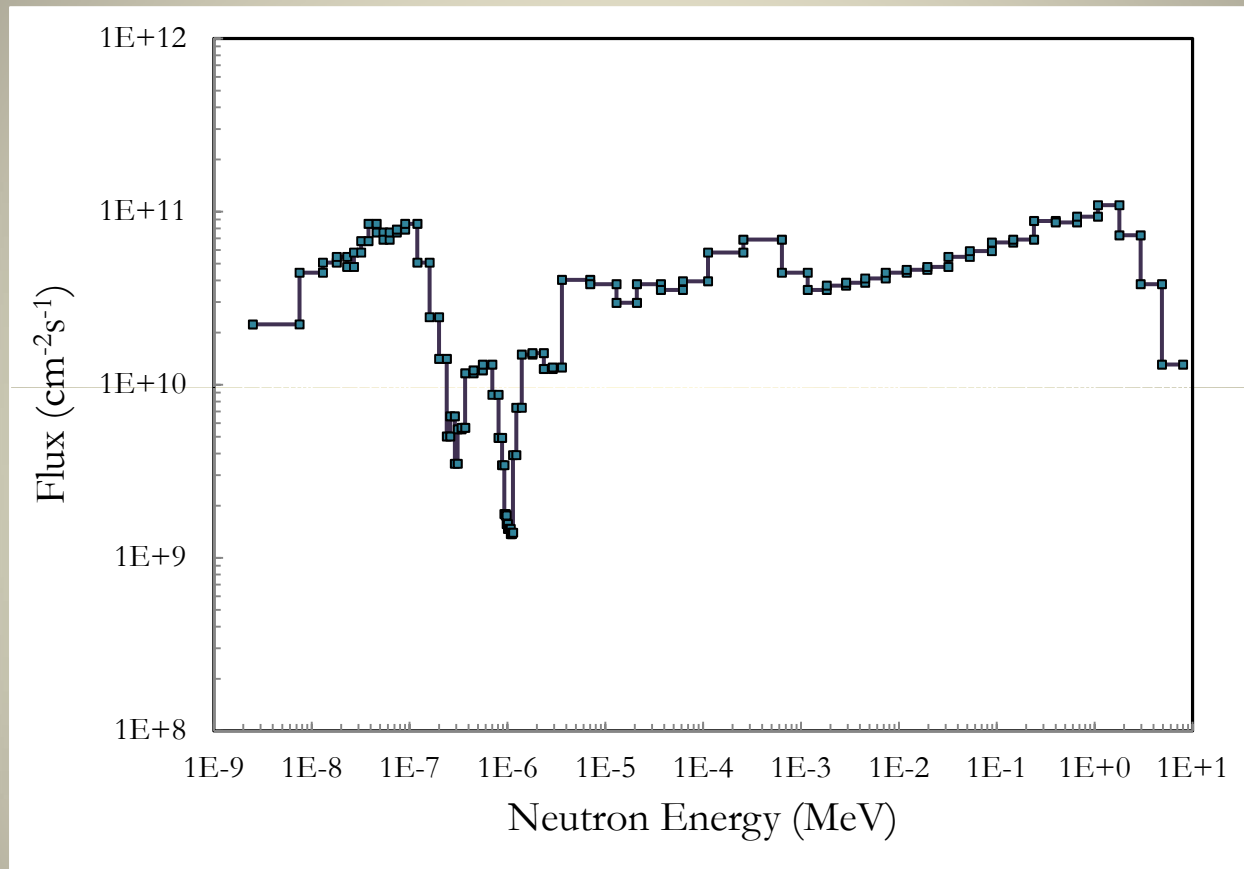


$S(\alpha, \beta)$ for Paraffin & Aqueous Solution



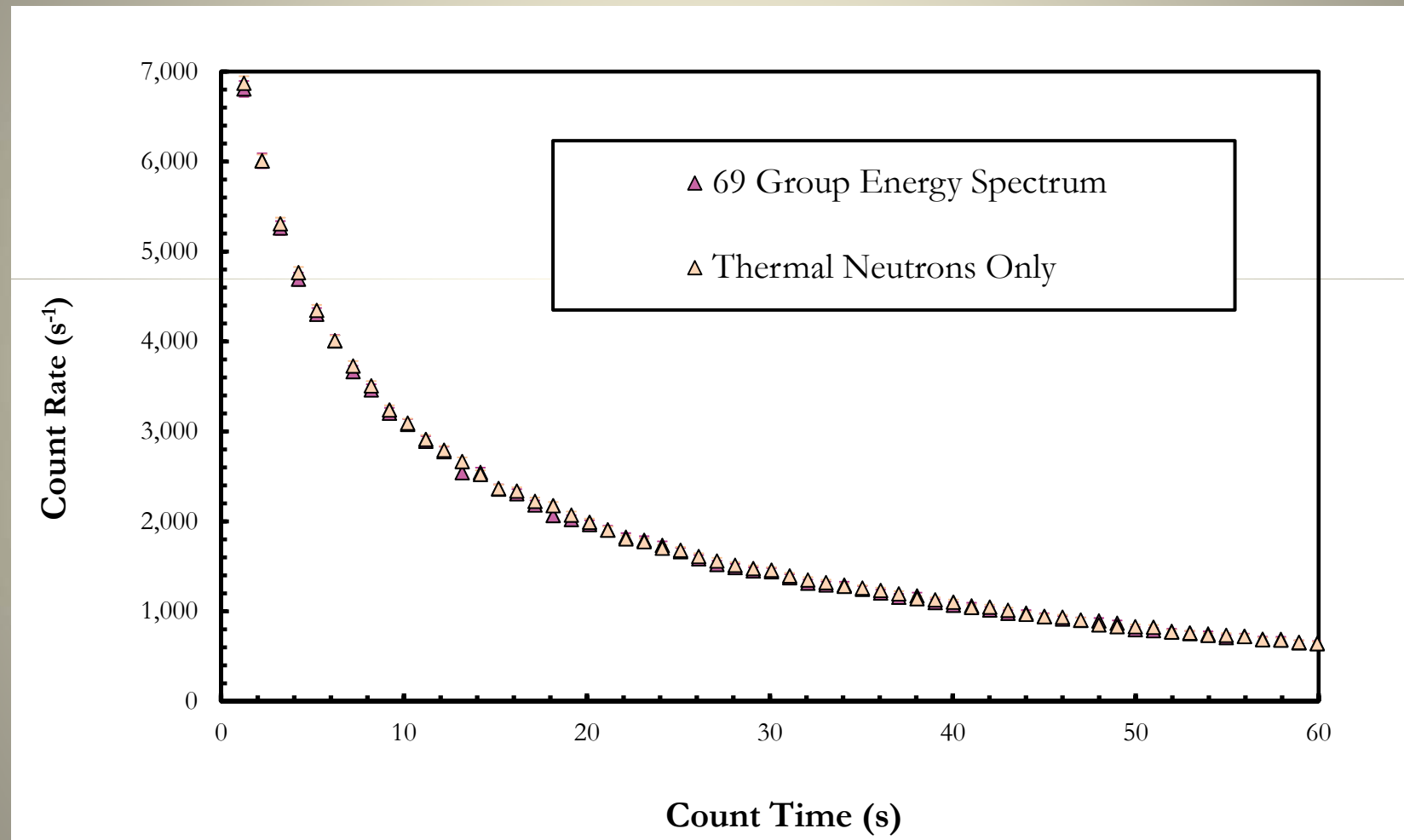
Molecular binding and crystalline effects are important at low neutron energies

Neutron Flux Estimate in SLOWPOKE-2

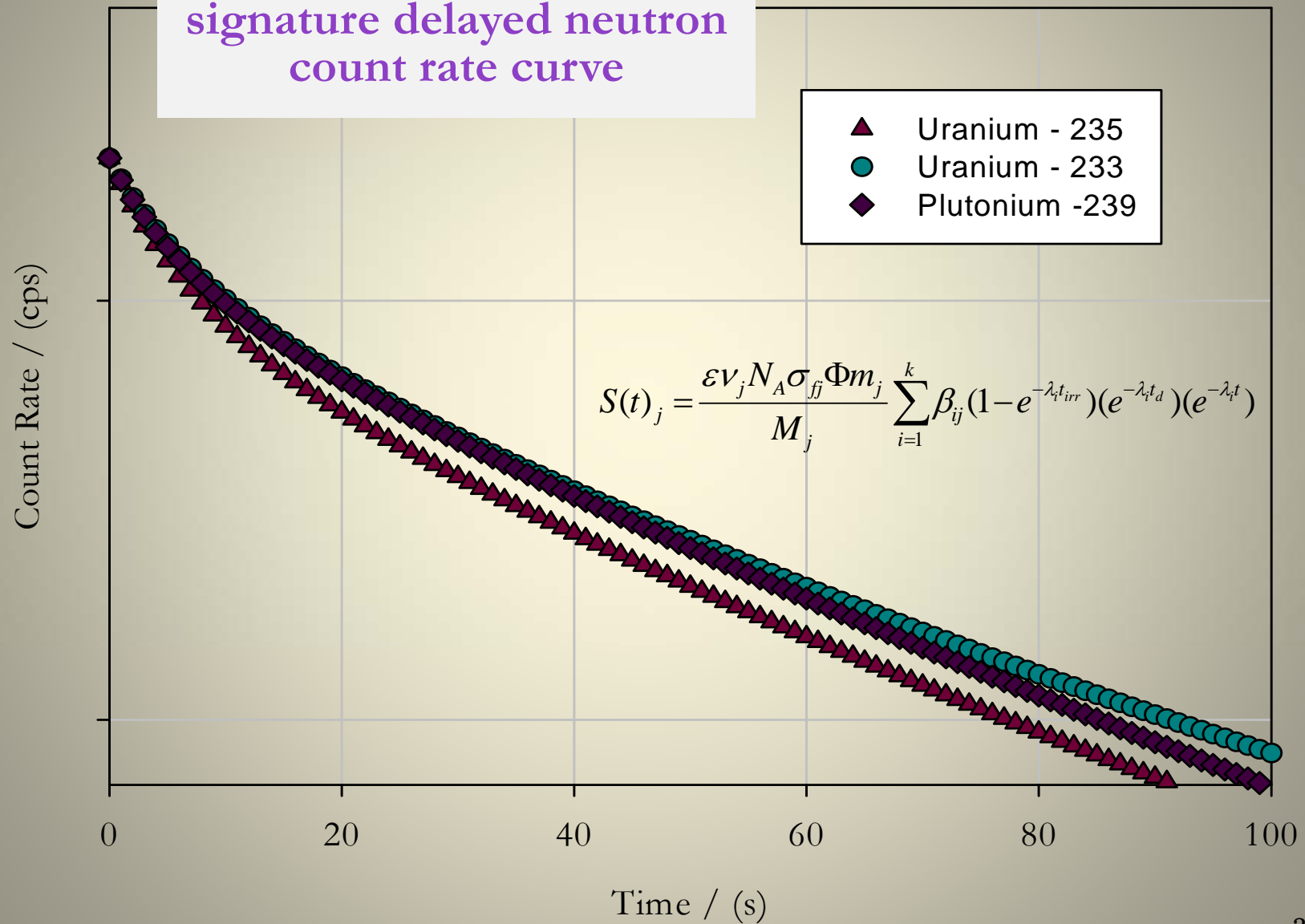


- 69 Group Energy Spectrum from K. Khattab, I. Sulieman, (2010, Syrian MNSR)
- Magnitude of flux determined experimentally in DNC irradiation site

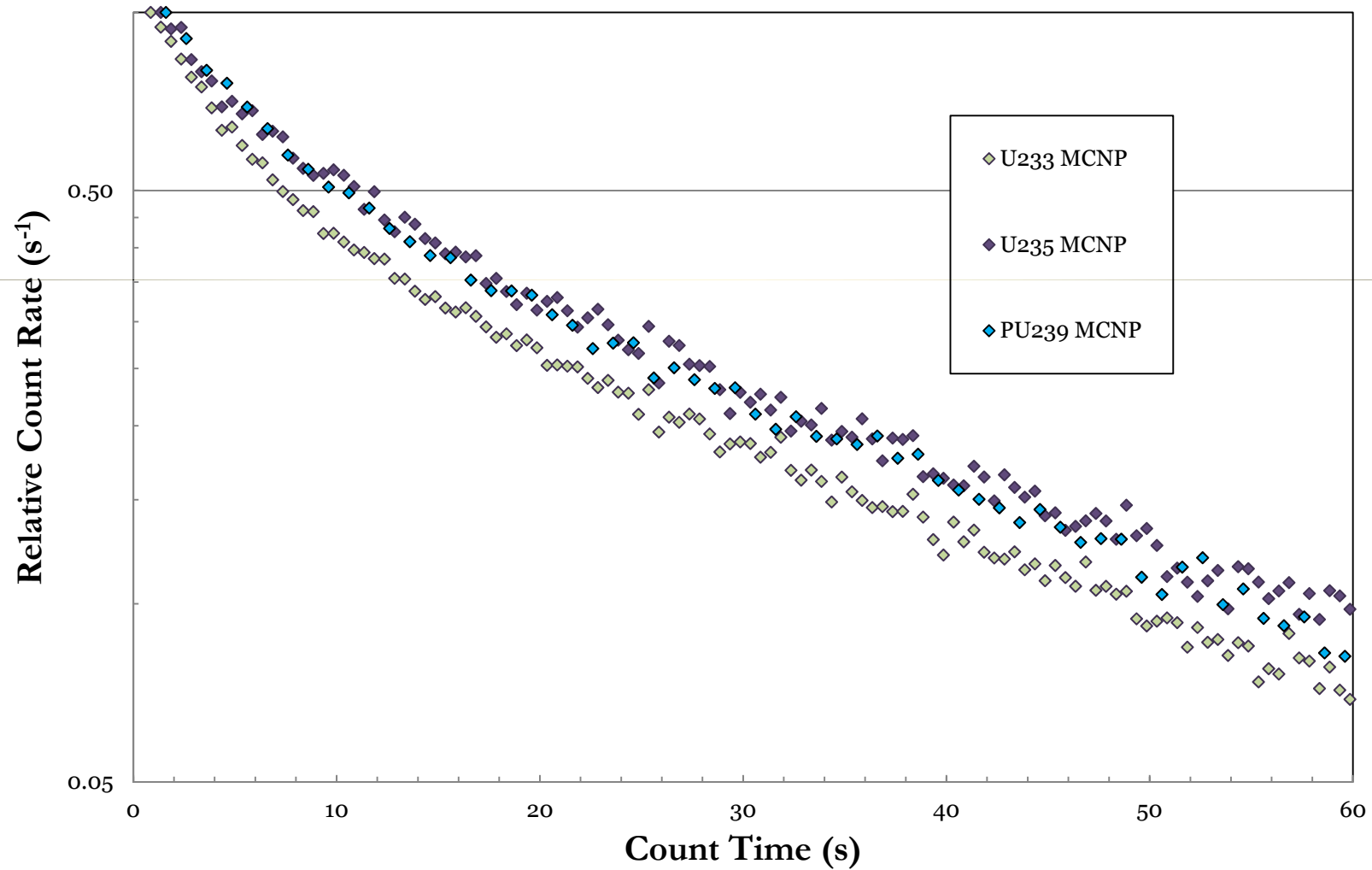
Neutron Flux & DN Behavior



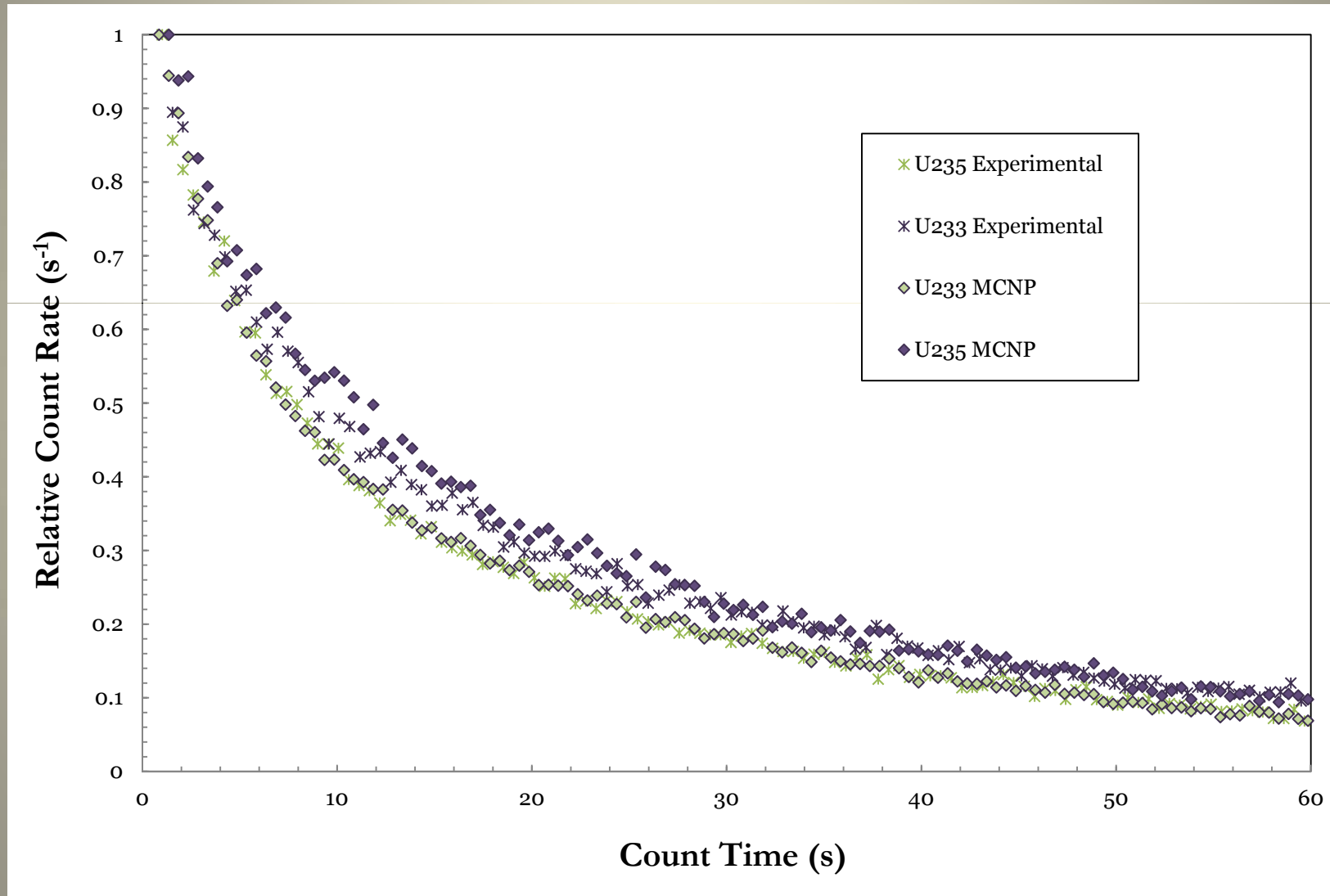
Each fissile isotope has a signature delayed neutron count rate curve



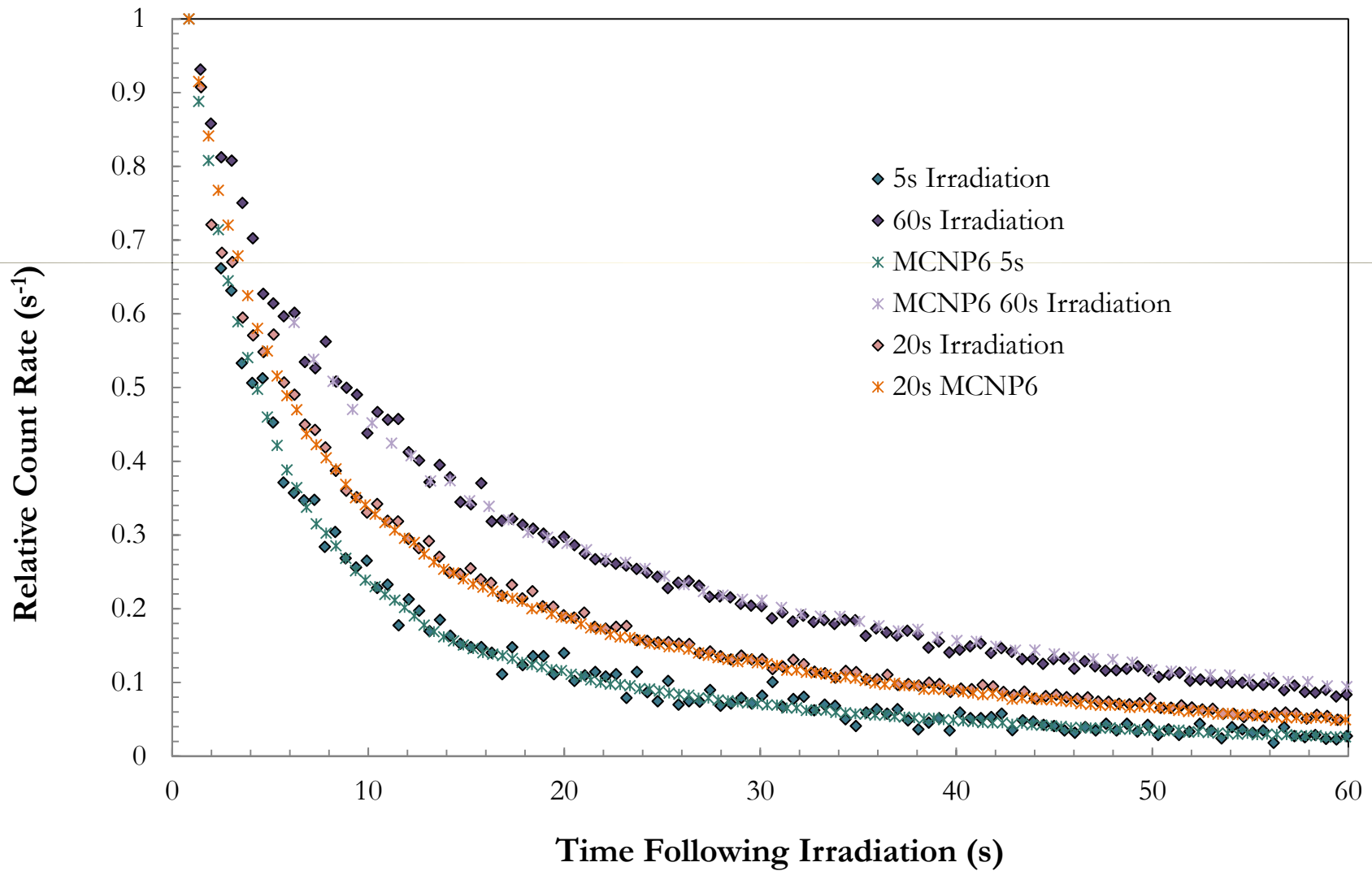
Temporal DN Behavior



Experimental Measurements & MCNP6



Irradiation Duration



CINDER vs. ACE Model

MCNP6 Initial Release Notes:

PHYS:N EMAX EMCNF IUNR **DNB unused* FISNU COILF CUTN**

DNB =delayed neutron control

-1001 Analog delayed neutrons from ACE tables (if available, otherwise from CINDER tables)

-101 Analog delayed neutrons from CINDER only

-1 Analog delayed neutrons from ACE tables only

0 No delayed neutrons produced

1-15 biased number of neutrons produced per fission.

101-115 Biased number of delayed neutrons from CINDER tables

1001-1015 Biased number of delayed neutrons from ACE tables, otherwise from CINDER tables

CINDER vs. ACE Model

MCNP6 Initial Release Notes:

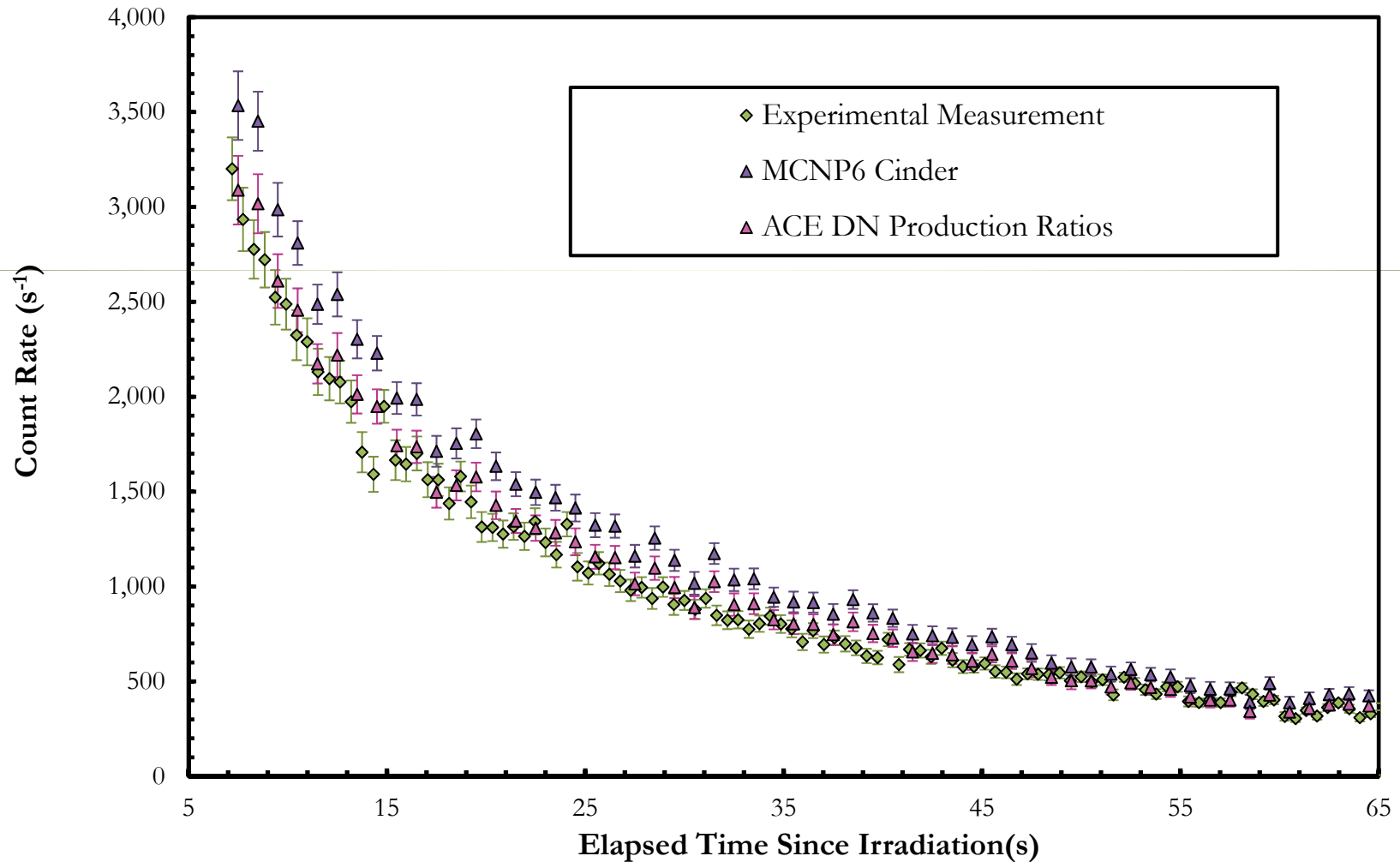
PHYS:N EMAX EMCNF IUNR **DNB unused* FISNU COILF CUTN**

-1001

-101

	ACE	CINDER	JEFF [*]
Prompt n	11992	11990	
Delayed n	60	96	
Weight Prompt n	1.52e-5	1.52e-5	1.58e-5
Weight Delayed n	7.61e-8	1.22e-7	1.06e-7

Un-normalized Comparisons – ^{235}U ($\pm 1\sigma$)

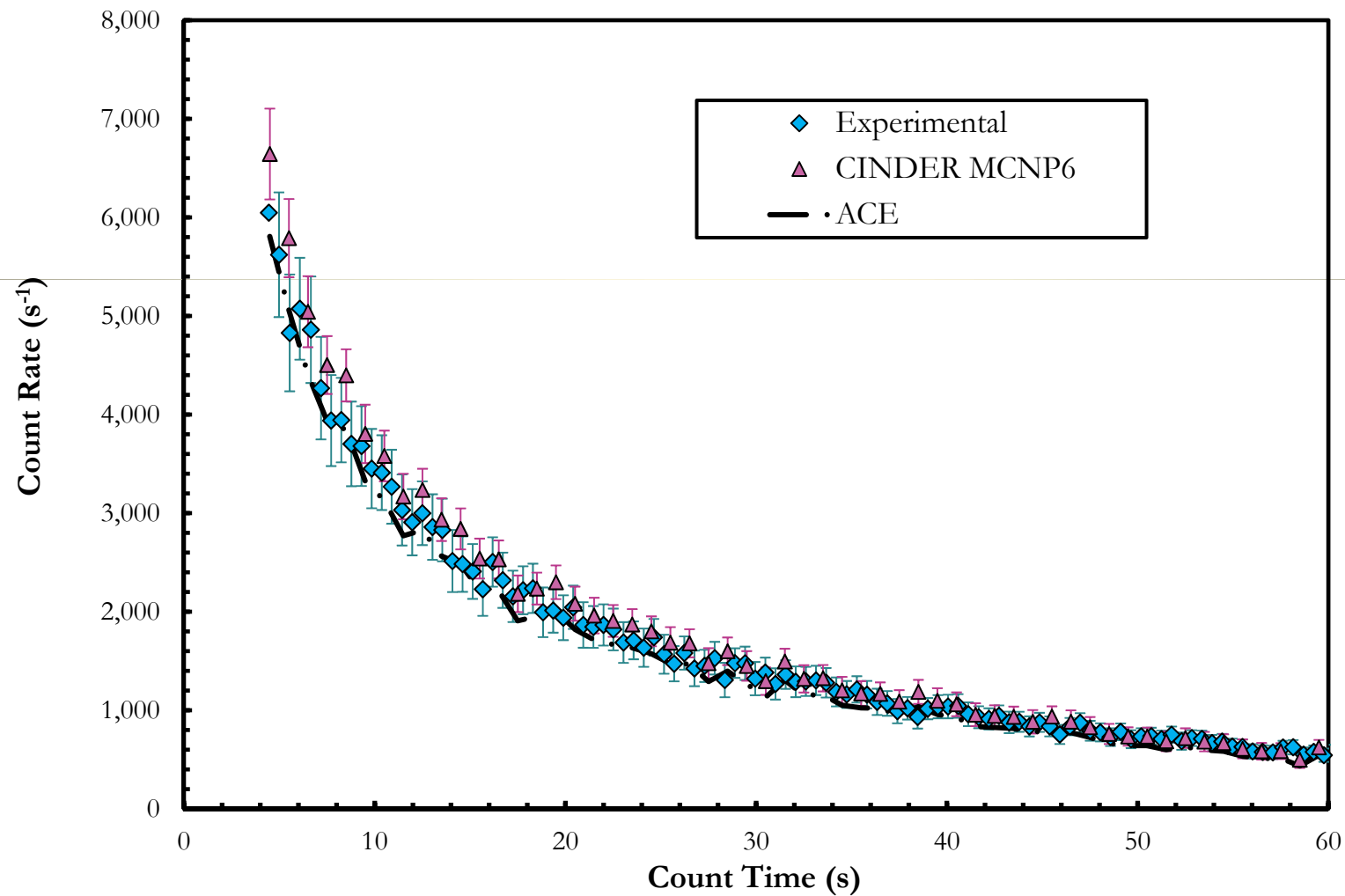


Future Work

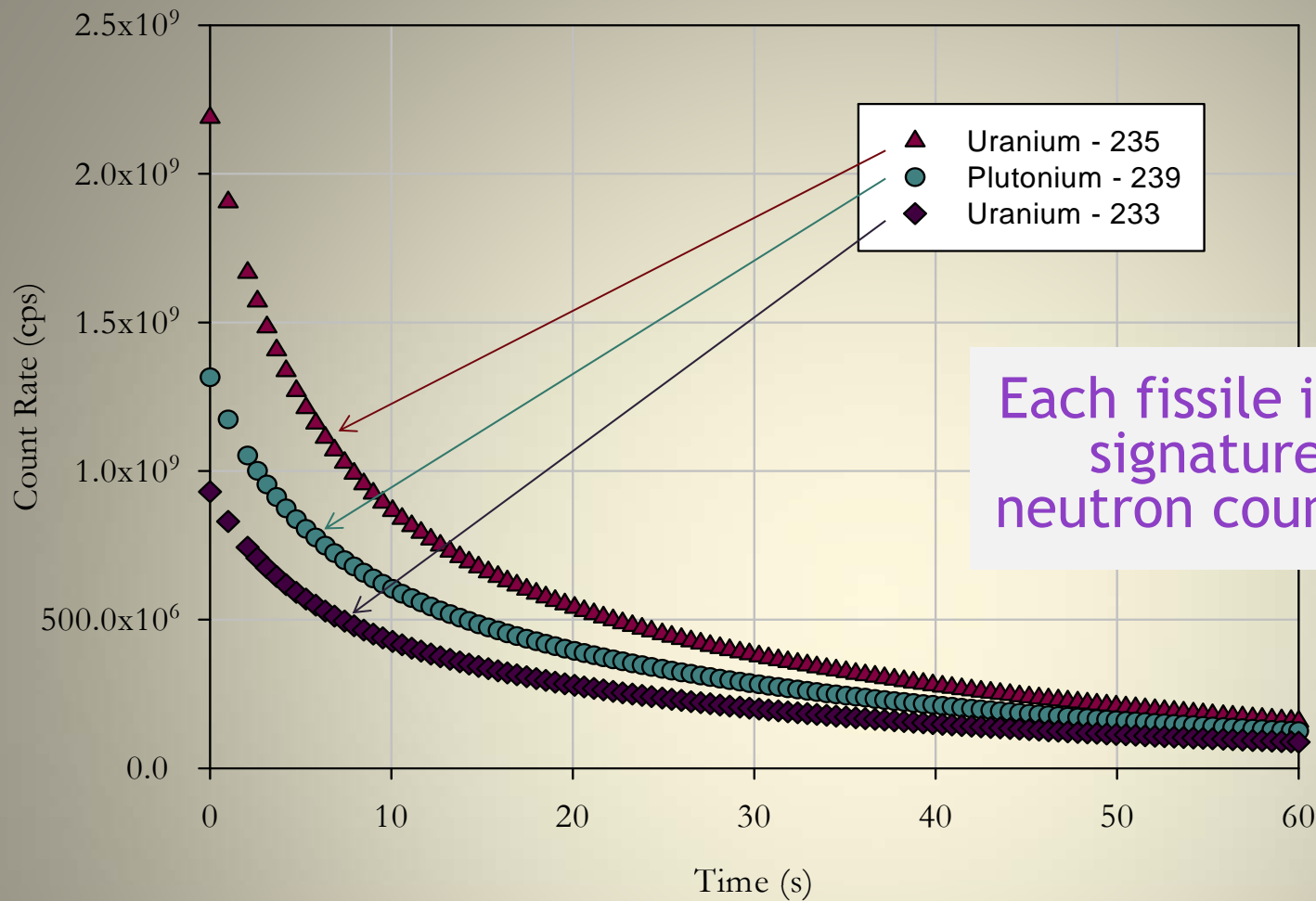
- Continued DNC experimentation and MCNP6 modeling
 - Longer Count Times
 - Plutonium-239 & Uranium-233
 - Mixtures of multiple fissile isotopes
- Delayed Gamma Emission from Special Nuclear Materials
 - System to be built at RMC late 2011/2012
 - To be modeled in MCNP
- Work to be presented and submitted to ANS Conference in 2012

Thank you

Un-normalized Comparisons – ^{235}U



Supplemental Information



Isotope j Count Rate:

$$S(t)_j = \frac{\varepsilon v_j N_A \sigma_{ff} \Phi m_j}{M_j} \sum_{i=1}^k \beta_{ij} (1 - e^{-\lambda_i t_{irr}}) (e^{-\lambda_i t_d}) (e^{-\lambda_i t})$$

ε = system efficiency v_j = delayed n production
 ϕ = neutron flux m_j = mass of isotope
 λ_i = lifetime of group i t_{irr} = irradiation time

n isotopes:

$$S(t) = \sum_{j=1}^n S(t)_j$$

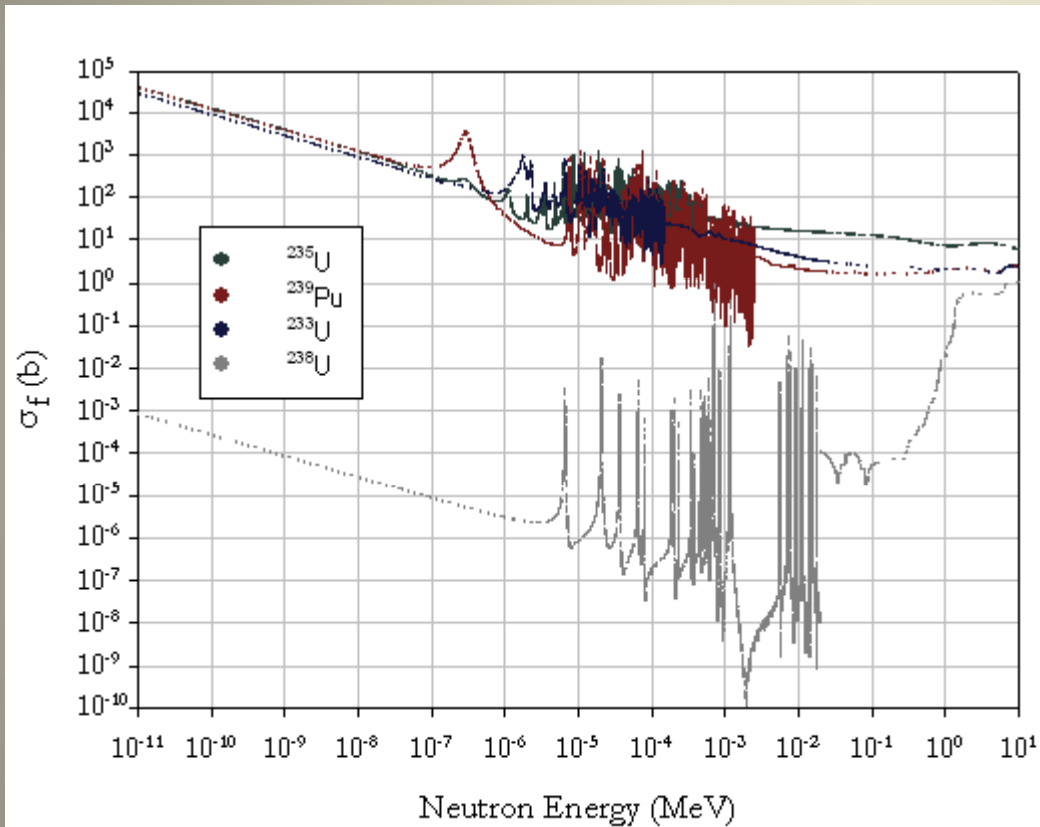
N_A = Avogadro's number
 M_j = Molar Mass of isotope
 t_d = decay time

Total Count Rate:

$$C(t) = S(t) + \text{Background}$$

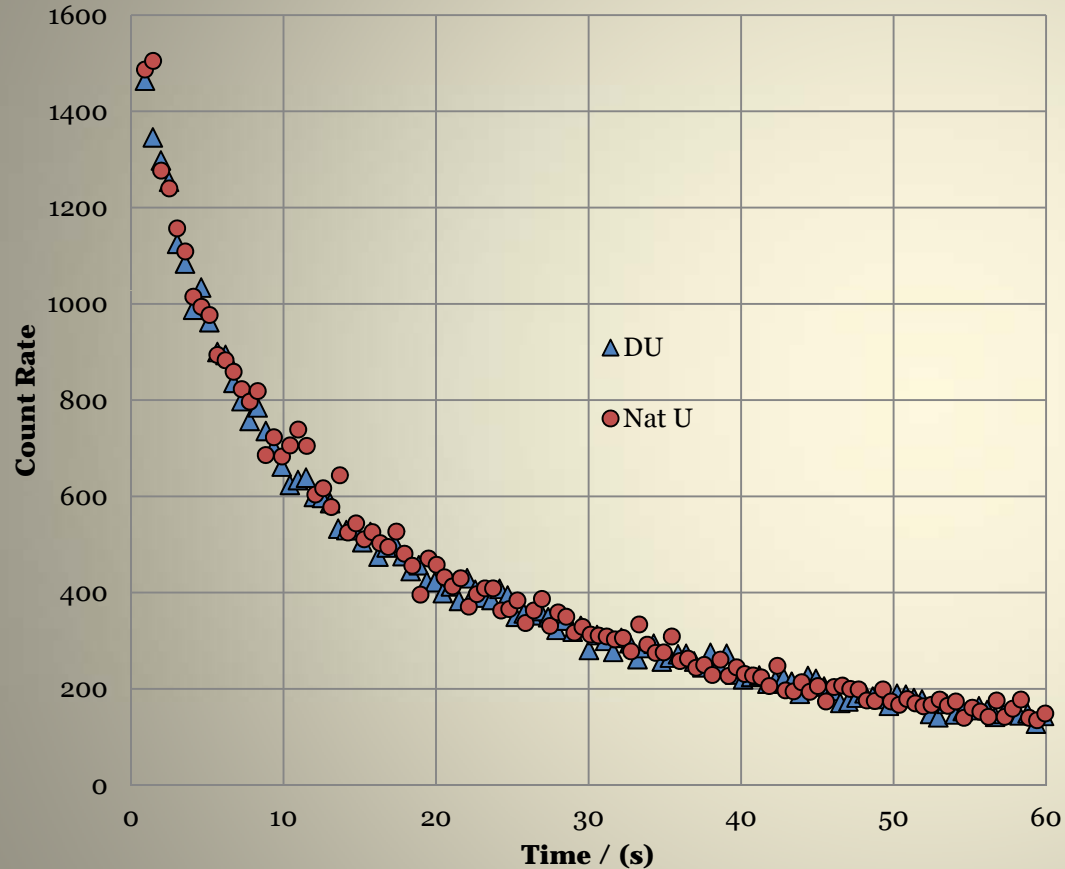
σ_f = fission cross section
 β_i = production ratio for group i
 t = count time

^{238}U Fission Contributions



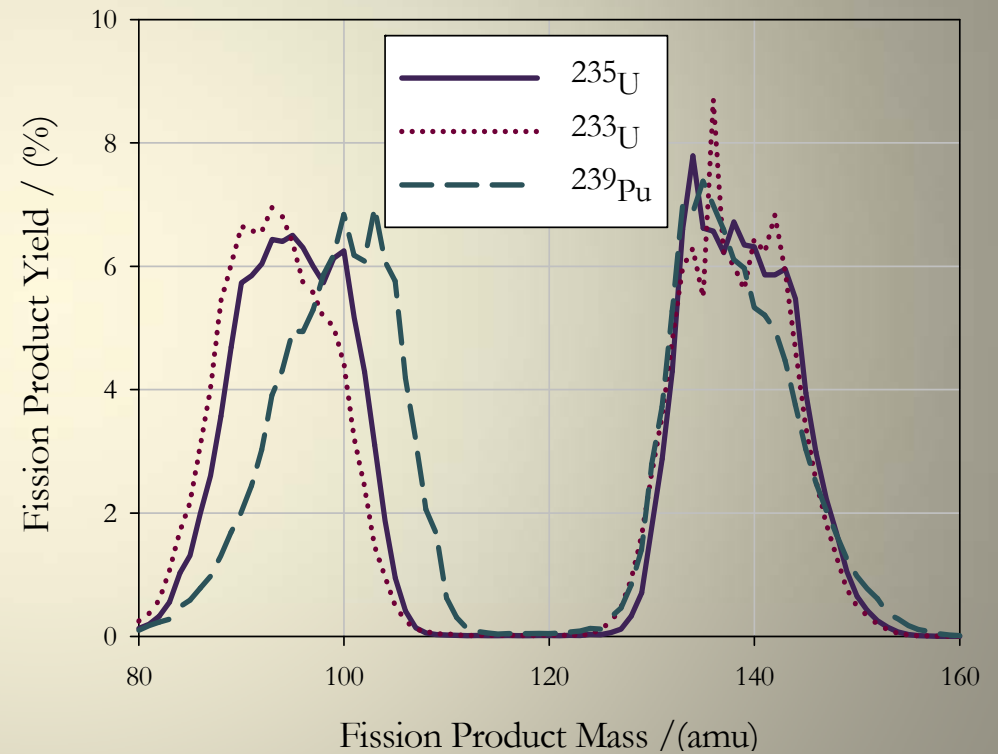
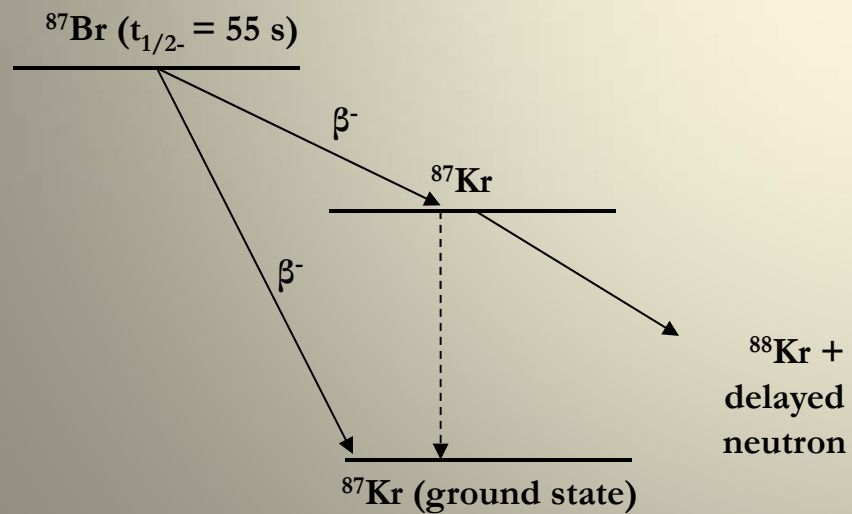
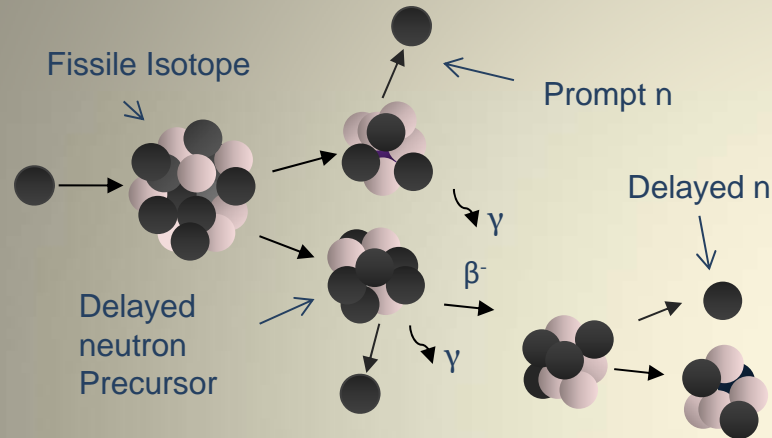
- Used epi/thermal ratio from Andrews thesis
- Account for flux, delayed neutron production, isotopic composition of U samples
- Determined that for natural U $< 0.001\%$ of DN recorded are from the fission of ^{238}U

^{238}U Fission Contributions

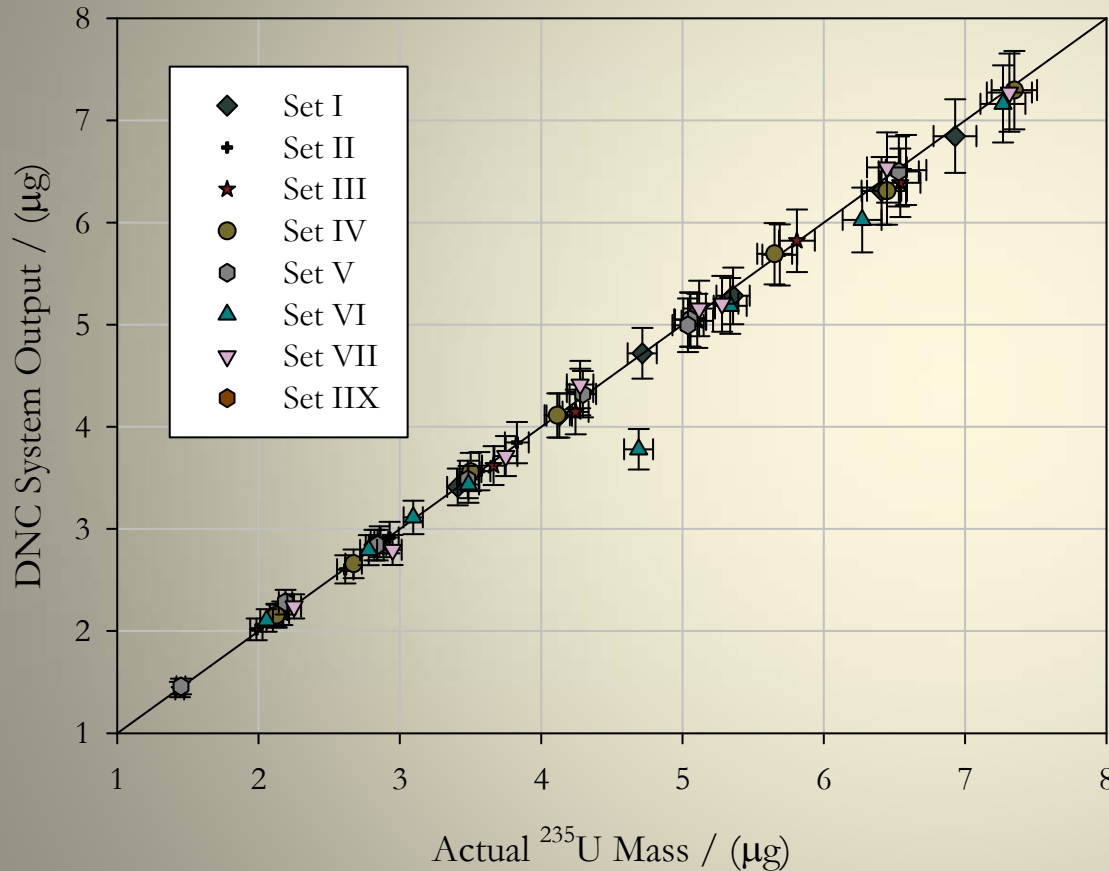


- Samples contain identical amounts of ^{235}U
- DU sample has $\sim 1.35\times$ the ratio of ^{238}U to ^{235}U
- Appears to have no effect on the magnitude of the count rate curve

Fission Fragment Yield & Delayed Neutron Precursors



System Reproducibility – Natural Uranium



Ideal: Slope = 1.0 Y-Intercept = 0.0

Set	Slope (2σ uncertainty)	Y-Intercept (μg) (2σ uncertainty)
I	0.98 ± 0.01	0.08 ± 0.04
II	0.96 ± 0.08	0.1 ± 0.4
III	1.00 ± 0.02	-0.02 ± 0.06
IV	1.0 ± 0.1	0.04 ± 0.06
V	0.98 ± 0.01	0.08 ± 0.04
VI	0.96 ± 0.08	0.1 ± 0.3
VII	1.00 ± 0.02	0.0 ± 0.1
IIX	0.99 ± 0.01	0.04 ± 0.06

System Validation – Depleted Uranium

Relative Error
-3.6%

Sample	Actual ²³⁵ U Mass (μg)	DNC System ²³⁵ U Determination (μg)	Relative Error
1	5.52 ± 0.06	5.3 ± 0.2	-3.45%
2	5.56 ± 0.06	5.3 ± 0.2	-5.35%
3	5.50 ± 0.06	5.3 ± 0.2	-3.01%
4	5.56 ± 0.06	5.3 ± 0.2	-4.32%
5	5.53 ± 0.06	5.3 ± 0.2	-4.78%
6	5.62 ± 0.06	5.4 ± 0.2	-4.45%
7	5.59 ± 0.06	5.4 ± 0.2	-4.24%
8	5.53 ± 0.06	5.3 ± 0.2	-4.47%

Actual ²³⁵ U mass (μg)	Total Solution Mass (g)	Experimental Mass (g)	Relative Error (%)
1.54 ± 0.04	0.290 ± 0.003	1.5 ± 0.1	-3.2
1.54 ± 0.04	0.379 ± 0.003	1.5 ± 0.1	-3.2
1.54 ± 0.04	0.568 ± 0.003	1.5 ± 0.1	-2.6
1.55 ± 0.04	0.660 ± 0.003	1.6 ± 0.1	-0.6
1.53 ± 0.04	0.819 ± 0.003	1.5 ± 0.1	-3.9
1.46 ± 0.04	0.738 ± 0.003	1.5 ± 0.1	-1.4
1.59 ± 0.04	0.943 ± 0.003	1.5 ± 0.1	-3.8

Error independent
of solution nitric
acid solution volume

Determining ^{235}U Content

